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Small power and heat generation systems on the basis of propulsion and innovative reactor technologies

Proceedings of an Advisory Group meeting held in Obninsk, Russian Federation, 20–24 July 1998

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FOREWORD

In the future for developing regions and remote areas one or two power reactors in the 50 MW(e) to 100 MW(e) range could be appropriately applied for electricity and heat production. Introducing and managing such a small programme with conventional reactor systems would require a mature supporting technological infrastructure and many skilled and highly trained staff at the site, which might be a problem for some countries. An increased number of small conventional reactors (e.g. with on-site refuelling) would increase the burden and expenditure for assuring security and non-proliferation. To this end, the time has come to develop an innovative small reactor concept that meets the following requirements: reliable, safe operation with a minimum of maintenance and supporting infrastructure, economic competitiveness with alternative energy sources available to the candidate sites, and significant improvements in proliferation resistance relative to existing reactor systems.

Successful resolution of such a problem requires a comprehensive systems approach that considers all aspects of manufacturing, transportation, operation, and ultimate disposal. Some elements of this approach have been used previously in the development of propulsion (ship and space) nuclear power systems, with consideration given to many diverse requirements such as highly autonomous operation for a long period of time, no planned maintenance, no on-site refuelling and ultimate disposition.

It is with this focus that the IAEA convened the Advisory Group on Propulsion Reactor Technologies for Civilian Applications in Obninsk, Russian Federation.

This meeting, which included participants from ten countries (Canada, China, Egypt, France, India, Indonesia, Japan, the Republic of Korea, the Russian Federation and the United States of America) brought together a group of international experts to review and assess the propulsion reactor design features and operational experience, mode of its alternative application, as well as to discuss the systems approach and requirements for innovative small reactors and rationale for selecting them.

The IAEA would like to express its thanks to all those who took part in the AGM, and particularly to the Institute of Physics and Power Engineering (IPPE) for hosting the meeting. Special thanks go to V. Chitaykin (IPPE, Obninsk, Russian Federation) and to S. Kazakov (JSC Malaya Energetika, Moscow) for assisting in the preparation of this publication.

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SUMMARY

1. INTRODUCTION

The Advisory Group Meeting (AGM) on "Propulsion Reactor Technologies for Civilian Applications" was held in Obninsk, Russian Federation, 20–24 July 1998. The AGM was convened by the International Atomic Energy Agency (IAEA) and was hosted by the Federal Scientific Centre, Institute of Physics and Power Engineering (IPPE).

This meeting was organized as a forum for experts of Member States to advise the IAEA on the different types of water and liquid metal cooled ship propulsion reactors, barge mounted power reactors and innovative reactor concepts which do not require on-site refuelling, and other similar reactor types presently in existence or under consideration in their countries. The purpose of the meeting was also to obtain advice from Member States on their needs and interests in the context of the IAEA's small and medium reactor programme.

Attendance at this AGM included thirty-five participants and observers from ten countries (Canada, China, Egypt, France, India, Indonesia, Japan, Republic of Korea, the Russian Federation and the United States of America). Sixteen presentations were made by the participants on a myriad of propulsion reactors design and operational experience, and alternative applications (floating and land-based), as well as on a technical approach for developing small nuclear systems for use in developing regions and remote areas. Each presentation was followed by general discussion and the AGM concluded with a round table evaluation of future small reactor technology requirements and exploration of areas for enhanced international co-operation.

2. SUMMARY OF TECHNICAL SESSIONS

2.1. Ship propulsion reactors: operational experience, new development and alternative applications

The Russian Federation possesses a powerful ice breaker transport fleet which offers a solution for important socio-economic tasks of the country's northern regions by maintaining year-round navigation along the Arctic sea route.

Progressive design-constructional solutions being perfected continuously during four decades of nuclear powered ships development in the Russian Federation and well proven technology of all components used in the marine nuclear reactors give grounds to recommend improved marine nuclear steam supply systems (NSSSs) of KLT-40 type as energy sources for heat and power co-generation plants and seawater desalination complexes, particularly as floating installations.

Co-generation stations are considered for deployment in the extreme north of Russia. Nuclear floating desalination complexes can be used for drinkable water production in coastal regions.

The first nuclear ice breaker was laid in 1956 and commissioned in 1959. This year (1999) becomes the 40th in the history of the Russian civil nuclear powered fleet, which consists of seven nuclear ice breakers and one transport (lighter carrier) ship. Nuclear powered ship reactor characteristics are presented in Table I.

All Russian nuclear powered ice breakers use a KLT-40 reactor type design, the safety features of which are as follows:

— Self-protection, self-regulation and self-limitation of power due to negative reactivity coefficients over the whole range of reactor parameter variation.
— Natural circulation through primary and secondary circuits ensuring heat removal from the core following the reactor shutdown.
— The weight of control rods (CR) and the energy accumulated in the compressed spring of the CR drives assure downward movement from any position at de-energization for

reactor shutdown. It also excludes ejection of absorber rods from the core if racks or drive casing loose their tightness.

— The large heat accumulation capability of the reactor gives the operator a large time margin to analyse emergencies and to organize accident management.
— Use of passive safety systems.
— Use of diverse safety systems and redundancy in safety system elements.
— Wide use of self-actuated devices for initiation of safety system operation, including reactor shut down, when the most important safety related parameters exceed their design limits.
— Use of the containment structure as a special system for keeping the core under water, and providing for passive heat removal from the reactor following primary circuit loss of integrity.

TABLE I. NUCLEAR POWER SHIP REACTOR CHARACTERISTICS

Name of ship	Year of commiss-ioning	No. of reactors	Power MW(th), max.	Working power, MW(th)	Core enrich-ment, %	U^{235} content of a core,kg	Core life years
			Ice breakers				
Arctica	1994						
Siberia	1997			2 × 120			
Russia	1985	2	2 × 171	= 240	~36	~200	~4a)
USSR	1989		=342	or			
Yamal	1992			1 × 171			
			Light carrier				
Northseaway	1988	1	1 × 135	~120	~36	~200	~4[a)]

a)All core replaced

The total operating record of the Russian propulsion nuclear reactors under extreme working conditions (rolling, vibrations, impacts of ice-floes, frequent manoeuvring) exceeds now 150 reactor-years, while that for the main equipment items on some operating reactors amounted to 125 000 h. Single failures occurred in individual components of the propulsion NSSSs, but no incidents associated with chain reaction control violation or inadmissible release of radioactivity occurred. The failures emerged, as a rule, following expiration of specified service life. Analysis of operating conditions and metallographic study of damaged elements in a primary circuit has shown that faults are governed by the following processes: thermocycle loads, corrosion wear, irradiation, and overpressurization. The main reasons for the thermocyclic effect are: operational transients (plant heating, cooling down, changes in power level, etc.), cyclic mass exchange between cold and hot coolant flows, disordered mixing of coolant flows with significantly different temperatures and non-steady pattern of coolant flow at heat exchange under significant temperature gradient conditions. Small thermocyclic cracks were revealed in pipes connecting a reactor with pressurizers, in internal headers for water supply from the purification system to reactor coolant pumps in internal shells of pumps' flow chambers, etc. The indicated defects became the reason for in-depth analytical and experimental studies of equipment operating conditions in the propulsion and test reactor plants, particularly thermal and strain parameters monitoring.

Some corrosion wear effect was associated, as a rule, with deterioration of a primary water chemistry condition (as compared to a specified one). In particular, the fact that steam generators were operated with excessive salt content in feed water, that could be the reason for failures of some steam-generating tubes and other items of SGs. As a result of irradiation of reactor internals, an increase in the force required to be applied to movement of reactivity control rods was observed by the end of the specified lifetime.

Due to systematic work on improvement of equipment and systems optimizations of their operating modes the reactor plant main equipment specified lifetime has been increased from (25–30) $\times 10^3$hrs for the first NSSSs up to (100–120) $\times 10^3$hrs for the modern ones. In order to further increase this performance indicator, extensive work is currently performed for the inspections of the. most loaded items in equipment and piping during the decommissioning of the first ice breaker reactor plant. In particular, samples are cut out from the reactor pressure vessel, the nozzle connecting the RPV to steam generator the pressurizer, heat exchanger, RPV closure head, pressurizing system pipes, etc. It is also planned to inspect items of the SG, RCP, CRDM and other key components. Metallographic tests of cutted-out samples and analysis of obtained results are currently proceeded. Based on the comprehensive inspection program a decision on extension of running equipment to the specified lifetime will be adopted, eventually up to the vessel hulls' service life.

Japan Atomic Energy Research Institute (JAERI) conducted feasibility studies on a concept of a compact light weight marine reactor for thermal power of 100 MW on the basis of conventional technologies with three types: an integral type reactor, a semi-integral type reactor, and an integral type reactor with a self-pressurized pressurizer. At the same time, JAERI surveyed cargo ships and tankers requirements for a nuclear powered system which could be competitive with the conventional engines of ships.

The studies revealed that all these reactor system concept designs need to become more compact and light weight For example, for the Japanese nuclear powered ship Mutsu, the radiation shielding occupied 70% of the weight of the whole reactor system and the secondary shielding 88% of the weight of the total shielding. As a result, it is one of the key design considerations for the compact and light weight reactor to make the radiation shielding component to be as compact as possible.

For the simplification of the reactor system, there is some room for improvement even in the land-based PWRs. The Japanese designers proposed a simplified PWR concept named JPSR featuring a full range self-power-controllability and a passive engineered-safety-features-system. They consider that the simplification of the reactor system is very effective in improving the reliability of the system, reduced required qualified manpower in operation and maintenance, improving safety and also reducing construction and operation cost. In order to simplify the system, the JPSR adopts a large pressurizer, canned pumps, passive residual heat removal system, and eliminates the functions of volume and boron concentration control system, seal water supply from the chemical and volume control system (CVCS), and so on.

In the marine reactor, the chemical shim reactor power control system is basically not adopted thus avoiding criticality due to seawater possibly entering in the case of the ship sinking. In this point, the marine reactor system is simpler than that of the present land-based PWR. However, necessities of the system simplification together with application of automatic operation system in the marine reactor are rather strong under the circumstance of a limited number of operators and the absence of support from land, which are more disadvantageous than for the land-based PWRs. These circumstances require the marine reactor system to have less possibility of accidents, to maintain core integrity and to remove decay heat by a simple and reliable way without manual operation in case of accident. That is, by some ingenious contrivance so that an accident has no potential to occur or by means of a passive engineered-safety system.

In order to create an advanced marine reactor system which is lightweight, compact, simple and safe, JAERI has developed an innovative integral type reactor system named MRX. Several new technologies are adopted in the MRX to suit the above mentioned proposals: a water filled containment, an in-vessel type control rod drive mechanism (CRDM), a passive decay heat removal system and so on. Feasibility of the MRX design concept for a commercial ship was evaluated by checking compatibility of the systems, and evaluating safety and economics in the whole system. New technologies adopted in the MRX such as the in-vessel type CRDM have been under development. The possible output range of the MRX is about 50 to 300 MW(th). The MRX can be used not only as a marine power source but also can serve multipurposes such as seawater desalination, heat supply, and electric power generation for distance district in floating and land-based modes.

Bhabha Atomic Research Centre (India) conducted feasibility studies for a 100 MW(th) pressurized water reactor which is being developed for cargo ship propulsion [1]. It is expected that the plant will generate a shaft horsepower of 29 000 and will be able to propel a ship, having a total displacement of 50 000 t, at a speed of about 17 knots. Apart from this, India has a long coastline and some of the coastal areas experience severe water shortage. Most of the coastal areas are away from coalmines and the regions have both water and electricity shortage. As India has very small oil and gas reserves, producing nuclear power in the region is a viable alternative both for power generation and seawater desalination. A preliminary feasibility study indicates that typically a ship with nuclear propulsion will weigh about 30 to 35% more than the conventional ship. This increased weight can adversely affect the cargo-carrying capacity. However, the study indicates that beyond a cruising radius of about 15 000 miles the amount of fuel which is required to be carried in a conventional ship will offset the disadvantage of higher plant weight in a nuclear ship. The study also indicates that a typical cost of such a plant is likely to be about 100 million US dollars.

Desirable features of India multipurpose reactor are as follows:

(i) Compactness and lower weight of plant. The components are arranged near the reactor pressure vessel to ensure compactness of design.
(ii) Higher system reliability. Each primary system loop is provided with two pumps. This will result in a relatively larger availability of primary system and thereby increase the reliability of basic heat transfer mode.
(iii) Larger core life. The core consists of 150 fuel assemblies arranged with a core edge of 1077 mm. Fuel assemblies are hexagonal in shape having a dimension of 63 mm across flat. Active core height is 1400 mm. For reactivity control, 19 control rods are provided. Preliminary study indicates that a fissile material inventory of about 140 kg should result in a core life of about 20 000 MW(d). Design is yet to be optimized for increasing the core life.
(iv) Reducing effect of accidents and restricting radioactivity release to a lower value. The containment vessel has a diameter of 8.7 m and a height of 10.5 m. The plant is provided in total with eight primary heat transport pumps. This will ensure primary water circulation, in the event that some of the pumps are not available. In addition the steam generator is placed at a higher elevation than the core. This will ensure water circulation by thermo-siphoning should complete power supply be lost. In the event of loss of coolant accident all the radioactive leakage can be contained in the containment vessel. This vessel will be designed to withstand the pressure build up, subsequent to such an accident.

It was pointed out that such a plant could also find use in of barge-mounted floating nuclear power plants.

The Institute of Nuclear Energy Technology, Tsinghua University, China conducted developments of a multipurpose small reactor (200 MW(th)). The reactor can be installed into a ship, and can be used for seawater desalination, electricity and heat production. The basic innovating features are as follows:

- Integrated arrangement, self-pressurization;
- Dual pressure vessel structure;
- Low operating temperature and pressure, and low power density reactor design which provided increased operating margins and improved fuel economy;
- Reactor core cooling with simple, passive means which use "natural" driving forces only;
- Adopting innovative hydraulic system to drive the control rods;

[1] At the meeting, these features were presented by CEA,Cadarache,but no paper was provided for publication in the proceedings.

- State of the art digital instrumentation and control systems and an advanced man-machine interface control room, console-type work stations, software controls and integrated, prioritized alarms and procedures;
- Innovative design of the plant arrangement and advanced construction concept adopted to minimize cost, to shorten the construction schedule and to meet safety, operational and maintenance criteria.

The Korea Atomic Energy Research Institute (KAERI) is carrying out the programme for the SMART (system-integrated modular advanced reactor) to be utilized in small scale electricity and heat generation and for barge-mounted application. The design combines well established commercial reactor design with new, advanced technology. A substantial part of the technology and design features of SMART has already been proven in industry and new innovative features will be proven through various tests. Despite disadvantages in the power production cost due to the small power output (~80 MW(e)), SMART can derive economic advantages from:

(a) a simplified system with a reduced number of pumps and valves, piping, instrumentation and wiring, etc.,
(b) flexible operation,
(c) modularized components,
(d) on-shop fabrication of components,
(e) better match to the grid, and
(f) low financial risk, etc. SMART is expected to fully satisfy the Korean as well as the international safety and licensing requirements.

CEA and TECHNICATOM (France) have about 200 reactor-years of experience from propulsion and small experimental reactors. Special design features for a future small reactor are formulated as follows:

(a) use of proven technology in which France has already had wide experience;
(b) integrated primary circuit operating at about 140°C and 10 bars, hence use of simpler and more reliable techniques and shorter construction times;
(c) introduction of passive features;
(d) suppression of soluble boron; and
(e) intrinsic simplicity of design and automatisms leading to reduced number of personnel for operation.

The small reactor is designed with low pressure and low primary coolant temperature in comparison with standard 900 MW(e) reactors. It was pointed out that lower primary temperature leads to:

(a) reduction of the fuel enrichment due to higher moderating ratio through increased density;
(b) possibility of high (60–70 GW·d/t) or very high burnups (<100 GW·d/t) due to considerably reduced corrosion and fission gas release through decreased temperature;
(c) reduced occupational exposure in the maintenance zones, and
(d) increased safety margins in accident situations. French estimations show that, with innovative architecture, systems simplifications and use of new technologies the overall kW·h costs of these SMRs could be of the same order of magnitude as large sized plants currently in operation.

2.2. Floating (barge-mounted) nuclear power plants: needs and status of development

The National Atomic Energy Agency (BATAN), Indonesia has concluded that to cope with the energy requirements in the remote islands and less developed regions of Indonesia small or very small nuclear reactors producing electricity and/or process heat could be appropriately applied. The barge mounted gas-turbine power plants, constructed in Surabaya shipyard, have been operating so far to provide temporary power for Balikpapan city. By size requirement, a nuclear propulsion reactor is the most attractive example so far envisionaged. Economic competitiveness despite the small size is the first issue to resolve. It is understood that small and very small power reactors have certain advantages in general (such as adaptation to small grid, less financing required, more manageable for domestic participation, etc.), but the cost of generated electricity per kW· h is usually high. While for the deployment at remote areas or at the less developed regions the main criterion for the small and very small reactor alternatives in principle remains, the least generating cost, additionally the following are considered:(a) the largest social gain, (b) zero or the least government subsidy and (c) a smaller cost to upgrade the infrastructure and transportation means in order to remove the "remoteness" qualification. These competitiveness criteria are applied, for both conventional financing, and for BOO (build-own-operate) or other non-conventional financing schemes (e.g. barter supplement). In principle an evaluation based on "profit and risk sharing" is considered.

The Russian State Concern for Electric and Thermal Energy Production of the Russian Federation Ministry for Atomic Energy plans to construct several floating nuclear power plants (FNPP) and the first one will be constructed in the North of Russia at Peveck, Chukotska Region. The FNPPP includes two KLT-40C reactor plants, electrical equipment complex, stand-by diesel and boiler plants and coastal infrastructure. An international competition was held by the Russian Nuclear Society for the best design of a small NPP. The twin KLT40C reactor FNPP won the first prize. Economic analysis has shown that for the extreme north of Russian a floating NPP on the basis of the ship propulsion reactor is 25–30% more economical than with alternative fossil-fuelled sources of energy. Two KLT-40C reactors in a rated mode of operation will generate ~75 MW(e). Spent fuel and radioactive waste are stored on board the FNPP. Thus, the autonomous operation period (operation without supplies replenishment) of the FNPP is determined by the capacity of spent fuel storage. The autonomous operation of the FNPP is ensured by four nuclear core sets and lasts ~15 years. After the lapse of this period the FNPP is to be towed to the dock for overhaul, fuel unloading and hull docking. Two overhauls and three operating cycles are planned. After the completion of the third cycle the FNPP is to be towed from the site to the premises of the specialised dock for decommissioning. The feasibility study for the construction of a FNPP with two KLT-40C reactors in Perek is completed. The industry is about to commence manufacture of the main equipment of the FNPP.

A Canadian company, CANDESAL Enterprises Ltd. evaluated an FNPP containing two KLT-40 reactors, as a source of electrical energy and waste heat for RO (reverse-osmosis) desalination. This company envisages implementation of the newest seawater desalination technology which allows a significant reduction in both the specific energy consumption for the desalination process and in the cost of drinkable water produced. The Russian MINATOM and CANDESAL developed a joint design of a FNPP desalination complex.

2.3. A new innovative approach to small reactor system design

Some AGM participants pointed out that small reactor concepts which are currently developed are based on downsizing today's large reactor technology. They consider that the infrastructure necessary to support conventional nuclear power development is very expensive, and beyond the resources of most developing countries. A new innovative approach to system design is to be used to reduce the need for such an infrastructure. The efforts should be focused on the development of the reactor systems with joint consideration of the overall fuel cycle, waste issues proliferation, simplified operation, simplified and minimised system maintenance. Several organizations presented their preliminary studies on innovative small reactor concepts.

The LLNL (USA) initiated an evaluation of an innovative concept of reactors and fuel cycles for developing countries. The leading criteria guiding this evaluation are:

(a) proliferation — resistance
(b) economic competitiveness with alternative energy sources available to the candidate sites;
(c) inherent safety and;
(d) ease of operation and maintenance.

Non-proliferating ingredients identified include lack of refuelling (no storage of either fresh or spent fuel) on-site and use of U-235 (enrichment <20wt. %) for fissile fuel. The first criterion may best be achieved if the core can be designed with one fuel load per life and the reactor can be delivered to the site pre-assembled and pre-fuelled. The Japanese 4 S (super safe small and simple) — sodium cooled fast reactor concept was the first selected for evaluation because it appeared to have the promise of achieving the no refuelling requirement. It was pointed out that one of the major challenges would be to accomplish a small reactor innovative goal with an economically viable system. Today's economic approach to nuclear power is through economies of scale. Some meeting participants consider that the small reactor economic problem could be resolved through economics of mass production, coupled with cost saving achieved from dramatically reduced on-site installation, operation and decommissioning costs, reduced site infrastructure requirements and simplified licensing to overcome the loss of economies of scale. Small systems with no on-site refuelling and replaceable reactor modules reduce proliferation concerns, and infrastructure burden due to the elimination of fresh and spent fuel storages, as well as technological operations with fuel on-site.

While many of these requirements and design features have, in principle, been met by space power systems, practical land or barge based systems that might meet this challenge will be orders of magnitude larger than space systems, but may need to be one to two orders of magnitude smaller than the current LWRs. It was pointed out that other complex systems such as a modern gas turbine or combined cycle plants meet many similar requirements including standardized factory fabrication, simple installation and startup, and highly automated operation

Principle safety objectives for innovative small reactor system have been identified:

- all operational and severe accidents to be considered in the design basis should be passively eliminated and not result in life-limiting damage of reactor system and radiological consequences to the on-site staff;
- the reactor design is to be provided core debris retention, its coolability and subcriticality;
- all postulated severe accidents considered in the beyond design basis should not result in loss of the use site due to contamination.

To achieve these objectives, an effective design must incorporate innovative solutions. The general question to be resolved is to create a reactor design with passive, inherent physical responses to achieve self-regulation, heat production and heat removal in balance. NPP's safety with an inherently controlled reactor could be less dependent on correct control action and less vulnerable to automatic control system fault and/or to operator or maintenance errors because the power would eventually passively adjust itself, thus minimizes the potential for human error initiation.

The 4S FR concept was proposed and developed by CRIEPI — Toshiba (Japan). The main features of reactor design are a tall thin reactor core (an equivalent diameter ~90 cm, length ~4 m) and axially moveable radial reflector for compensation of burnup reactivity. Its reference design of ~50 MW(e) was for 10 years of electric power output without refuelling and without the use of the safety rod for burn up reactivity control. Now, the CRIEPI — Toshiba team has proposed a new design variant for 24 years of full power operation without refuelling. It uses part of the reactivity worth of the safety-rod in addition to the radial reflector segments for burnup reactivity control.

It was pointed out that although most of the technologies used in the 4S are already proven or under development, further R&D work is required for some key technologies. These are criticality experiments of the metallic core with reflector, higher reliability of the reflector driving mechanism and fuel performance of a long fuel slug. A full-scale critical experiment is important to evaluate the calculated results such as reactivity coefficients and critical conditions. As neutron leakage is enhanced in the 4S core, the conventional calculation method is not sufficient to accurately predict the core characteristics. A critical experiment is thus the most urgent R&D item. All reactivity during plant operation is controlled just by moving the reflector without feedback control systems. Thus, fine movements of the reflector are required. The technologies proposed for this purpose are all new to the nuclear industry, but are tried and tested in other fields. Reliability experiments are needed. The feasibility of keeping a long metallic fuel slug in the core for a long time needs to be carefully examined. The preliminary assessment of creep deformation after ten years operation by the metallic fuel performance code shows about 5% deformation and is within the allowable value. However, the performance of longer fuel slugs must be demonstrated.

Studies of lead-bismuth fast reactors are being carried out in the Russian Federation organizations IPPE and GIDROPRESS, in which a great deal of experience has been accumulated in the course of the development and operation of submarine reactors cooled with lead-bismuth eutectic. However, bismuth is expensive and the resources are limited. It is possible that its use must be confined to special applications, such as small reactors or to a limited number of fast reactors. For this reason lead cooling is also being studied.

The advantages of lead-bismuth and lead cooling are high boiling temperatures and the relative inertness compared with sodium. The melting and boiling points of sodium are 98°C and 883°C. For lead-bismuth eutectic the figures are 123.5°C and 1670°C and for lead 327°C and 1740°C at atmospheric pressure. However, in the core of a lead-bismuth or lead cooled reactor the pressure of coolant above the core can increase the boiling point to about 2300°C. The boiling points are well above cladding failure temperatures. The specific heats per unit volume of lead-bismuth and lead are similar to those of sodium but the conductivities are about a factor of 4 smaller.

The γ radioactivity induced in the coolant is small so that access can be made to the coolant circuit after a shutdown period of about 24 hrs. However the production of α radioactive ^{210}Po from bismuth, and to a lesser extent from lead, poses problems because of its migration from the coolant to the cover gas and its formation of aerosols. ^{210}Po are volatile, so that any leakage from the cover gas poses a hazard to the plant operators and the environment. In the early stages of development, the formation of deposits of lead oxide and other impurities posed problems.

A careful control of the purity of the coolant is required to avoid the formation of such deposits. It was necessary to develop corrosion resistant steels and to pre-treat the surfaces of components and also to use special inhibitors in the lead-bismuth coolant. More extensive studies are required for lead coolant to demonstrate the corrosion-resistance of structural materials.

Two design concepts have been discussed, SVBR-75/100 and ANGSTREM. SVBR-75/100 is designed to produce 75–100 MW(e). The study explores the feasibility of designing an SVBR-like reactor core to operate for 10 years without refuelling. A transportable version of the reactor called ANGSTREM can produce 30 MW(th) or 6 MW(e) or a combination of heat and electricity. A version producing up to 25 MW(e) has also been discussed.

A study has also been made of a lead-bismuth cooled reactor operating with an open fuel cycle. The fuel volume fraction should be above 60% and should be a high density fuel. A large core (4 m diameter and 1.2 m high) is proposed. Following an initial loading with 10–15% enriched uranium fuel it is possible to move towards reloading with depleted uranium fuel. The burnup target is greater than 20%.

Lead-bismuth alloy is inflammable in air or water. In principle, a Pb-Bi cooled reactor would not have to have an intermediate circuit between the primary coolant and the steam, however it was said that there had been severe incidents with Pb-Bi cooled reactors due to water/steam leaks occurred in the submarine steam generators. As a result of one leak some areas of the reactor core were plugged by the products of water and Pu-Bi interaction, causing meltdown of the core. Therefore the elimination of the intermediate circuit between the primary coolant (Pb-Bi) and the water/steam is questionable and needs additional R&D.

2.4. Application of space reactor technology in gas industry and medicine

The design features and technology of very small Russian space propulsion reactors, which use a direct conversion of nuclear heat into electric energy, were discussed. An innovative thermoelectric plant to convert heat from fired natural gas into electricity was created by IPPE (Russia). Its main objectives is to ensure effective electrochemical rust protection of a large gas pipeline located in the remote arctic regions. One of the unique features of this system is the combination of the cathodic protection with a number of additional service possibilities, among them; the use of natural gas directly from the high pressure gas pipeline as a source of heat, the remote monitoring, etc. Starting in February 1998, the plant has been under field-testing.

A lively discussion took place on design and main parameters of a very small (~100 kW) reactor for neutron and neutron-capture therapy developed on the basis of space nuclear technology by

IPPE and the Radiological Centre of the Medical Academy (Russian Federation). For such a reactor the problem was to provide maximum emission of neutrons from the reflector surface, because the fraction of emitted neutrons per fission is an important parameter for reactor type selection.

3. CONCLUSIONS AND RECOMMENDATIONS

1. *Status and prospects of propulsion reactor (PR) applications.* The PRs for ice-breakers and ships have accumulated about 150 reactor-years of successful operation. Recent developments in the Russian Federation, Canada, China and other countries have demonstrated, that power reactors originally designed for ship propulsion could be used for electricity and heat generation. Use of proven PR technology and new developments on small reactor (SR) presents a broader nuclear power options to meet individual Member States' needs for land-based and floating SRs.
2. *The adaptation of PR technologies for SR requirements.* It was agreed that any PR proposed for electricity or heat generation will have to satisfy the general requirements and specifications of nuclear reactors as defined by international bodies such as the IAEA and by national safety authorities. It was agreed also that reactors mounted on barges will have to demonstrate that they meet safety criteria under specific conditions such as storms, sinking or tilting, which will have to be made by the plant owner. The owner should also clearly show that the propulsion reactor plant, provided as land-based or floating SR, has been sufficiently modified so that the buyer countries cannot reproduce the technology for non-civilian applications.
3. *The requirements for SRs deployment in developing countries.* Issues needing further deliberation and clarification include the following:

   3.1. Peculiarities in design of floating version

   - Compliance to international codes & standards in marine engineering and maritime rules.
   - Additional items in safety analysis, e.g. initiating events like collision, sinking, fuel recovery after sinking accident, extent of security ad safeguards implied in the design of plant and ports.

   3.2. Peculiarity in licensing

   - Licensability in the country of origin.
   - Extent of development testing and operational experience.
   - Scope of liabilities and insurance coverage.

   3.3. Peculiarity in economic assessment

   Apart from criteria for least generation cost and capital cost:

   - Cost of remoteness related to construction, O&M;
   - Cost of storage of fuel reserve related to longer operating periods in comparison with fossil fuel plant alternatives.

   3.4. Financing

   If the ownership of the reactor and fuel is intended to remain with the vendor, the BOO (build-own-operate) scheme is the most appropriate.

   3.5. Domestic participation

   Since the design of NSSS or the whole plant shall be compact and modularized, domestic participation in construction of the nuclear portions of the plant may be quite limited. Domestic participation foreseen in:

   - Research and development (still needed);
   - Design;

- Construction of on-land/auxiliaries, living compartments of large ship/barge maintenance.

4. New design principles, innovative architecture, and specific features that only small size can offer may lead to the development of this type of reactors which could be cost competitive with larger sized traditional nuclear reactors. The programmes for developing countries should be centered on a new innovative approach to system design to be based on experience gained over the past forty years with four types of nuclear technologies (LWR, LMR, HTGR and MSR).
5. The proliferation of nuclear energy in the future could supply many developing countries with one or two nuclear power plants. Managing even such small programme might lead to proliferation of fissile materials in the world. Time has, therefore, come to give a fresh look to the idea of reactors without on-sit refueling, and barge-mounted systems shipped to an internationally monitoring site for refurbishment, recycling and waste disposal.

OPERATIONAL EXPERIENCE WITH PROPULSION NUCLEAR PLANTS

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Abstract

Russia possesses a powerful icebreaker transport fleet which offers a solution for important socio-economic tasks of the country's northern regions by maintaining a year-round navigation along the Arctic Sea route. The total operating record of the propulsion nuclear reactors till now exceeds 150 reactor-years, their main equipment items operating life amounted to 120,000h. Progressive design-constructional solutions being perfected continuously during 40 years of nuclear-powered ships creation in Russia and well proven technology of all components used in the marine nuclear reactors give grounds to recommend marine NSSSs of KLT-40 type as energy sources for heat and power co-generation plants and sea water desalination complexes, particularly as floating installations. Co-generation stations are considered for deployment in the extreme north of Russia. Nuclear floating desalination complexes can be used for drinkable water production in coastal regions of Northern Africa, the Near East, India etc.

1. STEPS IN DEVELOPMENT OF TECHNOLOGY

Russia is a single country in the world which now operates a nuclear-powered icebreaker-transport fleet which offers a solution for vital social-economic tasks of the country's northern regions by maintaining a year-round navigation along the Arctic Sea route.

The first nuclear icebreaker "Lenin" - was laid in 1956 and commissioned in 1959 as a pilot commercial vessel registered to the Murmansk shipping company. Experience of its first sailing seasons (1959-1964) had shown significant advantages of icebreakers with nuclear propulsion plant in the Arctic, particularly their practically unrestricted sea endurance and ice breaking capability. [1].

Up to the date, three degeneration of nuclear propulsion plants have been created and gone through comprehensive verification by long-term exploitation under severe operating conditions (Table 1).

The first triple-reactor propulsion plant with a loop-type configuration had been operated during six years in the icebreaker "Lenin". By the end of that period an operating record of its main components did not exceed 25,000 h. The experience gained with that vessel had shown the expediency to construct a whole series of nuclear-powered icebreakers for extensive operations in the Arctic seas. With that aim a new twin-reactor propulsion plant of a modular type had been developed. Its pilot specimen was installed in the icebreaker "Lenin" instead of the first nuclear plant. Successful operation and high performance indicators of the plant allowed it to be recommended as a basis one for subsequent icebreakers "Aretica" (Fig. 1), "Sibir", "Russia", "Sovetsky Sojus" and "Yamal". In 1994 the next capital icebreaker "50th Anniversary of Victory" was launched, which is currently being completed the construction.

TABLE I MAIN CHARACTERISTICS OF PROPLUSION REACTOR

Data	Reactor Index				
	OK-150	OK-900	OK-900A	KLT-40	KLT-40M
1. Ship incorporating NSSS	ice-break "Lenin"	ice breaker "Lenin"	icebreakers "Lenin", Arctica" "Sibir", Rossiya", "Sovetsky Soyuz" "Yamal", "50 let Pobedy"	lighter carrier "Sevmorput"	icebreakers "Taimyr" Vaigach"
2. Displacement, tons	16000	15940	23500	62000	18620
3. Total shaft power, h.p.	44000	44000	75000	40000	48000
4. Number of reactors	3	2	2	1	1
5. Reactor power, nominal MWth	90	159	171	135	171
6. Operation life with one fuel load, days	500	1050	1050	1460	1120
7. Primary coolant parameters at nominal power:					
1) core inlet temperature, °C	261	278	273	279	273
2) core outlet temperature, °C	284	318	316	311	316
3) primary system nominal pressure, MPa	18	13	13	13	13
8. Secondary circuit parameters at nominal power:					
1) steam flow, t/hr;	3x90	2x220	2x240	215	240
2) steam flow, °C	290	305	290	290	300
3) steam pressure, MPa	3.1	3.19	3.34	4	3.4

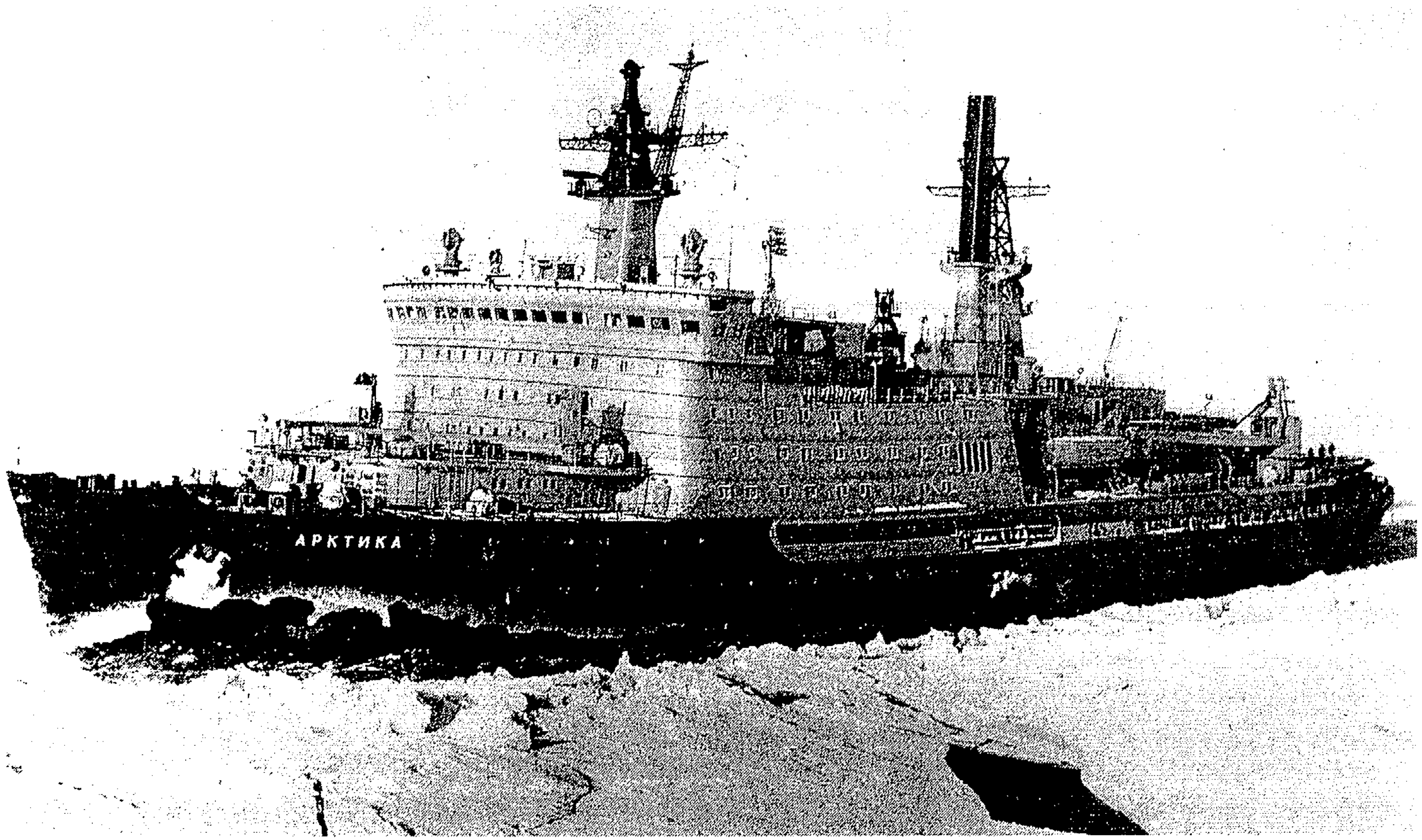

Fig.1. Nuclear icebreaker "Arctica" conducting transport ships on the Arctic sea route.

In spite of similarity of nuclear steam supply systems (NSSS) used for those icebreakers they were gradually step-by-step improved from one vessel to other. Their modifications were primarily directed to elimination of drawbacks revealed during operation of individual components and systems under harsh operating conditions in predecessor NSSSS, to enhancement of operational safety, technical & economic characteristics and operating life.

Safety of the reactor plants is relied upon the reactors inherent self-protection features and application of successive protective barriers, redundant systems and equipment, efficient safety systems.

The next stage in creation of propulsion NSSS was development of a single reactor plant for lighter-carrier "Sevmorput" and limited-drought icebreakers "Taymir" and "Vaygach", constructed jointly by Finnish and Russian enterprises. A decision to build the vessels with single reactor plant was justified by high operational reliability of this type NSSS.

The NSSS for lighter-carrier "Sevmorput" was designed with regard to requirements of the Code for safety of nuclear merchant ships developed by International Maritime Organization (IMO).

A number of unique operations have been performed in the Arctic by the nuclear-powered icebreakers. For instance, in 1977-78 vessel "Lenin" provided for the first time a year-round transport operations in the Arctic Sea route ways. In 1977 vessel "Arctica" carried out a voyage to the North Pole in active sailing, that consequently transformed into regular tourist cruises. In 1978 vessel "Sibir" carried out an experimental high-latitude sailing by a shortest way from Murmansk to Magadan.

In 1989 vessel "Sevmorput" carried out its first commercial voyage over the route: Odessa-Vietnam-Vladivostock, during which the ship's reactor plant way inspected by Administrations of host parts for conformity with requirements of the IMO Code document. For years of the nuclear-powered civil ships operation an appropriate infrastructure has been created in Russia, including specialised repair-technological enterprises, training Centre, auxiliary maintenance ships for the reactor refuelling.

3. RESULTS OF OPERATION

This year (1998) becomes the 39th one in the history of the Russian civil nuclear-powered fleet, which consists of seven nuclear icebreakers and one transport (lighter carrier) ship (Fig. 1). The fleet's forefather, the icebreaker "Lenin", is already removed from operation, while the construction of the newest icebreaker 50th Anniversary of Victory is nearing completion now. The main performance indicators of the nuclear-powered ships over the period from 1970 to 1997 are given in Table 2.

The total operating record of the propulsion nuclear reactors under extreme working conditions (rolling, vibrations, impacts of ice-floes, frequent manoeuvring) exceeds now 150 reactor-years, while that for the main equipment items on some operating reactors attained 120,000 h. During that period, no incidents associated with chain reaction control violation or inadmissible release of radioactivity were indicated.

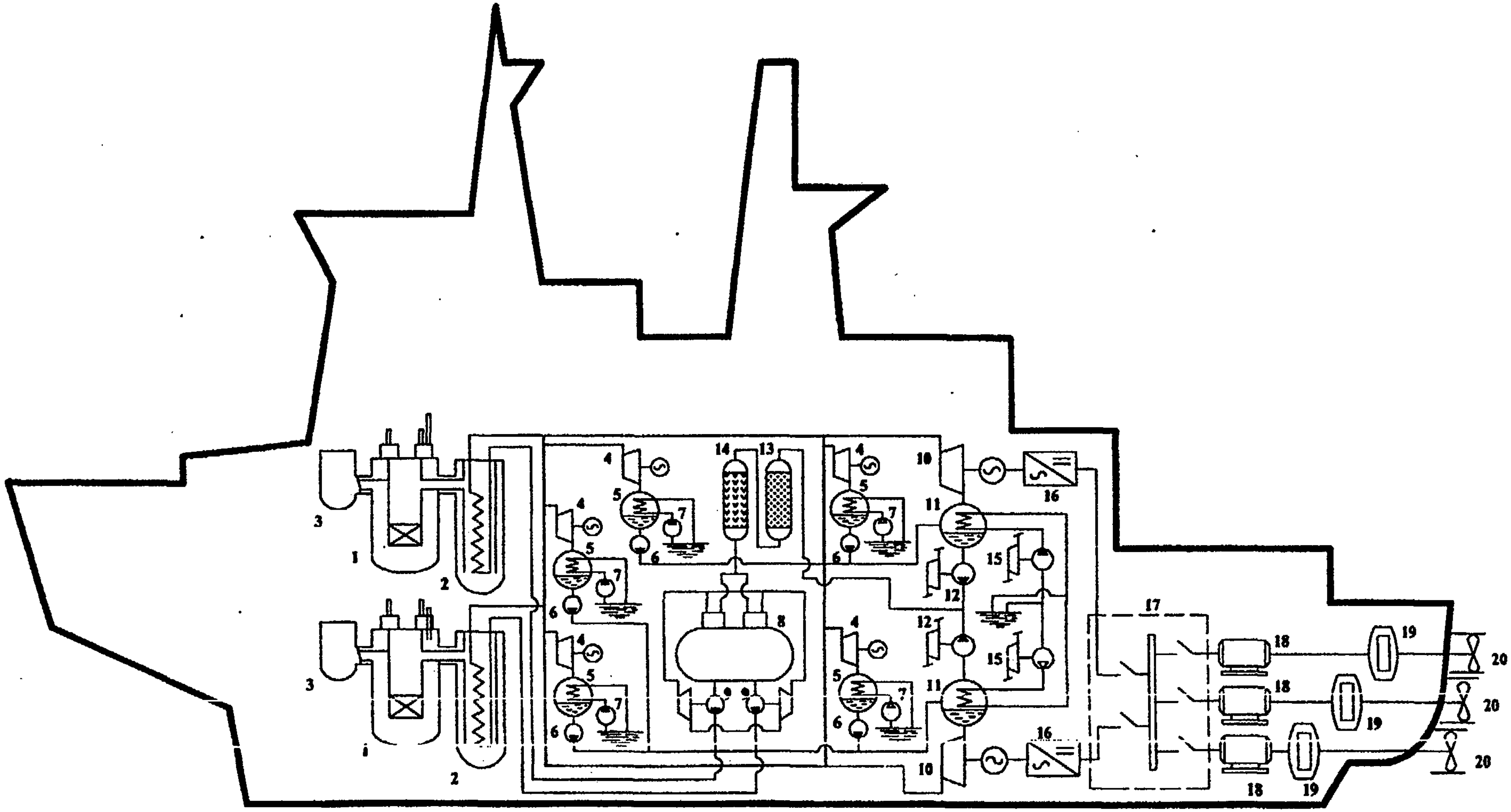

FIG. 2. "Arctica"-type icebreaker's nuclear power plant configuration
1-reactor, 2-steam generator, 3-reactor coolant pump, 4-auxiliary turbine generator (ATG), 5-ATG condenser, 6- ATG condensate pump, 7-deaerator, 8-turbine-driven feedpumps, 10-main turbine generator, 11-main condenser (MC), 12-MC turbine-driven condensate pump, 13-mechanical filter, 14-ion-exchange filter, 15-MC turbine-driven circulating pump, 16-converter, 17-propulsion motor power station, 18-propulsion motor, 19-deadwood bearing, 20-propeller

TABLE II OPERATION EXPEREINCE OF PROPULSION REACTORS

Characteristics	Name of ship									
	i.-br. 'Lenin'	i.-br. 'Arctica'	i.-br. 'Sibir'	i.-br. 'Russia'	i.-br. 'Sov. Souz'	i.-br. 'Tyamir'	i.-br. 'Vaygach'	i.-br. 'Yamal'	l.-cr. 'Sev-morput"	i.-br. '50th Anniver-sary of Victory'
1. Year of commissioning	1970	1974	1977	1985	1989	1989	1990	1992	1988	under con-struction
2. Averaged duration per year, days	230	225	232	258	232	255	252		291	
3. Total reactor operating record , h	106740* 106350**	123575 124276	94785 94043	60572 60011	52596 52239	57627	48305	35704 35081	59453	
4. Total energy produced , 10^3MWh	6523 6398	8091 7500	6095 6933	4294 4472	3254 3433	4495	4153	2602 2601	3378	
5. Distance sailed, miles										
1) total	654400	825177	740786	391761	305857	331995	213421	229819	232650	
2) incl. Through ice	560600	721644	472787	361283	258341	308134	196712	204648	77191	
6. Number of ships conducted	3700	2830	1711	1121	425	919	576	572		

i.-br. = ice breaker, l.-cr. = lighter carrier
* for reactor No. 1, ** for rector No.2

However, single failures were taken place in individual equipment pieces of the propulsion NSSSS, including small leaks of primary coolant into the environs and interfacing systems. The failures emerged as a rule following expiration of specified service life. Analysis of operating conditions and metallographic study of damaged elements in a primary circuit have shown that faults are governed by the following processes:

- thermocyclic loads;
- corrosion wear;
- irradiation;
- overpressurization.

Main reasons of thermocyclic effect are:

- operational transients (plant heating, cooling down, changes in power level, etc.); cyclic mass exchange between cold and hot coolant flows;
- disordered mixing of coolant flows with significantly different temperatures;
- non-steady pattern of coolant flow at heat exchange under significant temperature gradient conditions.

Thermocyclic cracks were revealed in pipes connecting a reactor with pressurizers, in internal headers for water supply from purification system to reactor coolant pumps, in internal shells of pumps' flow chambers, etc. The indicated defects became the reason for in-depth analytical and experimental studies of equipment operating conditions in the propulsion and test reactor plants, particularly thermal and strain parameters monitoring.

Corrosion wears effect associated, as a rule, with deterioration of a primary water chemistry regime (as compared to a specified one). In particular, the steam generators were operated with excessive salt content in feed water that could be the reason of failures of some steam-generating tubes and other items of SGs.

Resulting from irradiation of reactor internals increase in forces to be applied for movement of reactivity control members was observed by the end of a specified lifetime.

The operating experience has shown that abrupt changes in a primary pressure can be the reason of damages in internal structures made as semi-closed plena which walls were not designed for the primary coolant pressure. That was the reason of failure of a partition in the reactor coolant pump of icebreaker "Arctica".

Guides on optimal running of operational transients were developed for improvement of equipment operating conditions, and some modification was introduced in the equipment structures.

4. DESIGNER SUPERVISORY ACTIVITY, INSPECTIONS, LIFE EXTENSION

During the entire period of the NSSSs operation a supervisory activity has been carried out by the design organization as concerned reactor-related equipment and systems. Each failure or mis-operation event was thoroughly analysed, and recommendations for their elimination were then developed. Remarks of the plants personnel on operation of individual equipment items and systems were analysed and quickly removed; new improved arrangements were tested; the plants operating modes were optimised, etc. Inspections of equipment were carried out, weak points were identified and then removed in both operating and newly designed NSSS. Resulting from the supervisory activity actual operation models were determined, according which more precise calculations were carried out aiming at damage parameters corrections and residual service life prediction.

Due to systematic activity on improvement of equipment and systems, optimizations of their operating modes the reactor plant main equipment specified lifetime has been increased from (25-30)x 10^3hrs for the first NSSSs up to (100-120)x 10^3hrs for the modern ones.

In order to further increase this performance indicator extensive work is currently performed for the inspections of most loaded items in equipment and piping of the decommissioned icebreaker "Lenin" reactor plant. Particularly, samples are cut out from the reactor pressure vessel, nozzle connecting the RPV to steam generator, the pressurizer, heat exchanger, RPV closure head, pressurizing system pipes, etc. It is also planned to inspect items of the SG, RCP, CRDM and other key components. Metallographic tests of cutted-out samples and analysis of obtained results are currently proceeded. Resulting from the comprehensive inspection program fulfillment a decision on extension of running equipment specified lifetime will be adopted, eventually up to the vessel hulls service life.

Presently, the reactor cores are nearing their specified energy production potential of 2.1-2.3TWh. There is a prospect to increase this value, and consequently to improve an economic effectiveness of the reactor plants operation.

Long-term monitoring of the environment (sampling and laboratory tests of snow, soil, air, vegetation, etc.) in the area of the nuclear-powered ship homeport has no revealed any radiological effects of the ships. Obtained data are on the background level. Average annual dose to individuals of the ship personnel does not exceed 5mSv.

Available safeguard barriers and confinement systems practically completely eliminate radioactivity release, even in the most severe design accidents.

The Russian nuclear-powered ships are operated by highly qualified and experienced specialists, which along with their direct duties perform significant scope of research work aiming at enhancement of the reactor plants safety.

Actual level of nuclear and environmental safety of the latest modifications of NSSSs meets modern requirements of domestic and international regulatory documents, and hence removes restrictions on the nuclear-powered ships deployment. It is also demonstrated be the commercial cruises of the icebreakers with passengers.

5. PROSPECTS OF NUCLEAR PROPULSION PLANTS UTILIZATION

The application of progressive solutions for constructional and technological provisions, continuous perfection of safety systems with an Account of updated regulatory requirements for nuclear safety, and improvement of equipment items on the basis of operational experience feedbacks give grounds to recommend the propulsion nuclear reactors as advanced heat sources for co-generation power stations and sea water desalination complexes [2].

Operations along the Arctic Sea route seem to be problematic without vigorous work of the nuclear icebreakers. They provide pilotage of up to **80%** of vessels there. Therefore the decommissioned nuclear icebreaker should be replaced for new ones.

The far north and similar remote regions of Russian occupy more than half of the country's territory where the major part of mineral and energy resources are located, including oil, natural gas, nickel, gold, diamonds and rare metals. However, most places in these regions are not provided with a centralised power supply, have no fuel-energy resources expedient for effective utilization, and while fossil fuel delivery there entails great difficulties and expenses. For these regions, the application of NPPS, especially floating ones, becomes justifiable and prospective.

Creation of nuclear-powered icebreakers, cargo vessels and floating NPPs with standardized reactor plants for servicing Russia's northern regions represents a promising and economically efficient task, since the available integrated maintenance infrastructure can be used for new ships maintenance too.

Creation of floating power-desalination complexes is another prospective application of the propulsion reactor plants. It is a fact that drinkable water production has become an acute problem for many regions of the world, e.g., North Africa, the Near East, India etc. Drinkable water is also expensive in northern regions of Russia.

Among a number of different projects, IAEA is considering floating power desalination complexes, in particular those with the marine nuclear reactor plant of KLT-40 type and desalination facilities of distillation and reverse-osmosis types. The appropriate distillation facilities in Russia are designed by the Sverd NII ChimMash Institute (Ecaterinburg), while reverse-osmosis facilities are produced in particular by the Canadian firm "Candesal Inc".

In 1995 Memorandum on mutual understanding in the development of a desalination complex was signed by "Candesal Inc." and MinAtom of Russia (it was prolonged in 1997). By present a draft prospect on joint power-desalination complex is prepared [3].

High reliability and safety levels of the reactor plants are verified by their long operation experience. Projects of nuclear co-generation stations and power desalination complex based on the propulsion-type reactor plants represent a certain commercial interest and can be attractive for domestic and foreign investors.

6. CONCLUSIONS

(i) Powerful icebreaker - transport nuclear-powered fleet and adequate maintenance infrastructure have been created in Russia by the date.

(ii) Unique multiyear operation experience has been accumulated with the propulsion nuclear reactor plants, and their operational safety and reliability are convincingly confirmed.

(iii) Causes of rare typical failures of equipment and systems have been identified, and proper countermeasures were undertaken for their complete or partial neutralisation.

(iv) The reactor plants have gone through multistage evolutionary modernisation, so that their equipment and systems have now a long specified lifetime.

The KLT-40 NSSSs are still attractive for fast realization in new icebreakers, cargo vessels, floating heat and power co-generation stations and power desalination complexes.

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A MULTI-PURPOSE REACTOR

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Abstract

A integrated Natural circulation self pressurized reactor can be used for sea water desalination, electrogeneration, ship propulsion and district or process heating. The reactor can be used for ship propulsion because it has following advantages: It is a integrated reactor. Whole primary loop is included in a size limited pressure vessel. For a 200MW reactor the diameter of the pressure vessel is about 5m. It is convenient to arranged on a ship; Hydraulic driving facility of control rods is used on the reactor. It notably decreases the height of the reactor. For ship propulsion, smaller diameter and smaller height are important. Besides these, the operation reliability of the reactor is enough high, because there is no rotational machine (for example, circulating pump) in safety systems. Reactor systems are simple. There are no emergency water injection system and boron concentration regulating system. These features for ship propulsion reactor are valuable. Design of the reactor is on the base of existed demonstration district heating reactor design. The mechanic design principles are the same. But boiling is introduced in the reactor core. Several variants to use the reactor as a movable seawatwer desalination plant are presented in the paper. When the sea water desalination plant is working to produce fresh water, the reactor can supply electricity at the same time to the local electricity network. Some analyses for comprehensive application of the reactor have been done. Main features and parameters of the small (Thermopower 200MW) reactor are given in the paper.

1. A CONCEPT OF USING THE HEATING REACTOR FOR PROPULSION

Originally, the purpose of the heating reactor is to supply heat for Buildings in cities. Now we want to extended its application to propulsion, heat-electro-cogeneration, sea water desalination and refrigeration (for air condition) etc.

In the regions, where fresh water resources are deficient, transport of fresh water is need. Usually ship transport is used. This way to transport water is not convenient. If a reactor is mounted on a ship, as a desalination power source, at the same time as a propulsion power resource, that would be a interesting solution.

For example, in China for several islands fresh water is need. A movable desalination plant is a feasible solution. When the plant arrived the island, it can supply some additional electro- power to the local network. This isn't negligible in China.

Heat and power cogeneration (for sea water desalination and propulsion) will improve the economic features of the reactor.

(1). Multi- purpose ,extends applications of the Heating Reactor.

(2). As a power source, the reactor can work whole year, From view point of economy, it much better than only supply heat for buildings in cities.

2. HEATING REACTOR IS APPROPRIATE AS A PROPULSION POWER RESOURCE

As a propulsion power source Heating Reactor has following advantages:

(1) Sizes are limited. Heating Reactor adopted integrated arrangement. Whole primary loop is included in a pressure vessel. Sizes of the pressure vessel are limited. A compact arrangement for the ship propulsion power source is a important factor.

(2) Height is less. Hydraulic driving facility is used in the Heating Reactor. It makes the Height of the Heating Reactor much less than ordinal power reactor. This feature is beneficial for its application on ships.

(3) High reliability. In main safety systems of the Heating Reactor, there are no rotational components (for example, pumps). This feature improves the reliability of the reactor. As a propulsion reactor, it may work faraway from land base, so the reliability is important.

(4) Simple systems. Because of higher level inherent safety, there is no safety class electro-power source,. As well as high pressure and low pressure injection systems.

Burnable poison is adopted in the reactor, so boron concentration control and regulation system is need no longer.

(5) High Safety. Following features make the reactor more safe during its operation: double pressure vessel, natural circulation (There is no big sized penetration on the pressure vessel), self pressurizer, gravity boron injection and residual heat removing system with natural circulation etc.

This kind of reactor is proper for operation far away from land bases.

3. DESIGN SOLUTIONS

3.1 Design principles

(1) Reactor design will be Improved, but only on the base of existed Heating Reactor design, in order to minimize R & D work. (The Heating Reactor Sections, Systems and parameters are shown in Fig.1-3 and Tab. 1.)

TABLE 1 MAIN PARAMETERS OF THE NHR-200

Reactor power	200MWt	Core power density	36.2kw/L
Coolant pressure	2.5MPa	Height of the core	1.9m
Coolant temperature	210/145℃	Core eq. diameter	~ 1.9m
Pressure of secondary		Number of heat exchangers	6
loop	3.0MPa	Length of the H/Exg.	6.5m
Temperature of 2nd loop	145/95	RPV diameter	5.0m
Temperature of the heat grid	130/80℃	Height	13.6m

(2) Main safety features will be kept: including low pressure, low temperature, low power density and inherent safety systems.

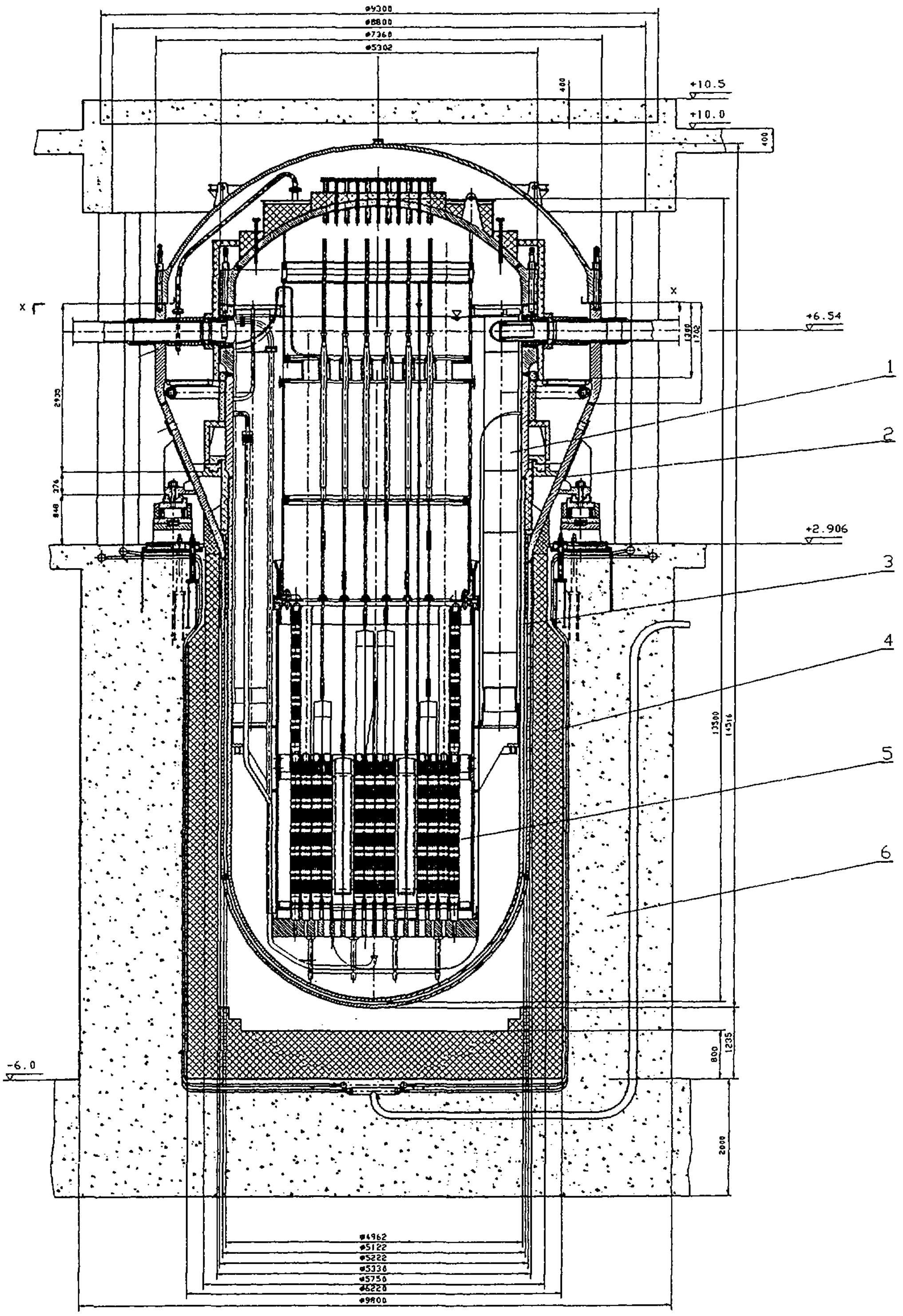
3.2 Main design variants

(1) Back pressure type power and heat cogeneration plant.

Using the heat, generated from the Heating Reactor to drive a back pressure turbine. Then the steam goes to the sea water desalination plant to produce fresh water.

The temperature at the outlet of the reactor core is 213℃.the in let temperature is 154℃, in the case of fresh water productivity=$1.2 \times 10^5 m^3$/day. The power can be used for propulsion is about 12000KW.

Main parameters of the cogeneration plant are as follows (see Tab. 2):



1-main heat exchanger; 2-containment; 3-RPV;
4-thermal isolation; 5-reactor core; 6-biological shield.

Fig. 1 Section of 200MW heating reactor

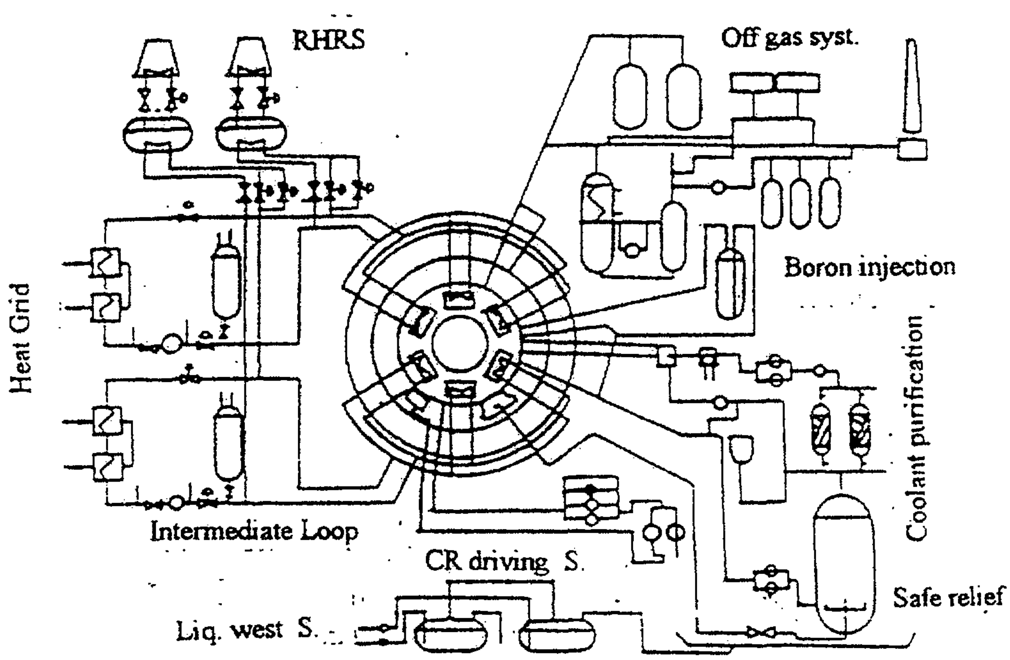


Fig. 2 Schematic System Diagram

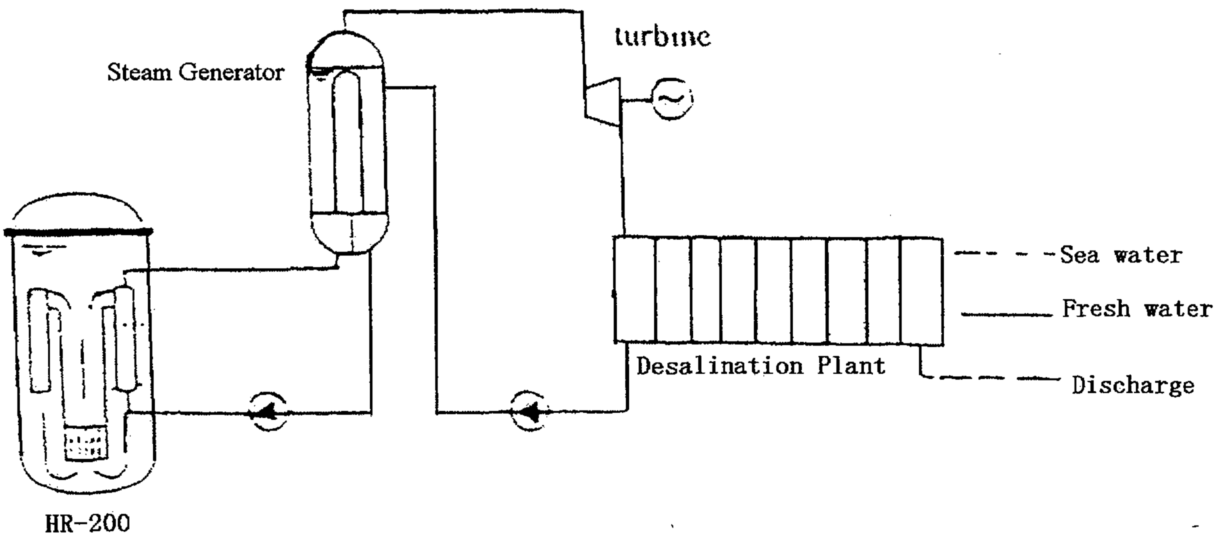


Fig. 3 Back Pressure Cogeneration Plant

TABLE 2. MAIN PARAMETERS OF THE BACK PRESSURE TYPE POWER-DESALINATION PLANT

Reactor power	200MWt	Intermediate loop water temperature	170/144℃
Electro-power	12MWe	Steam temperature	140℃
Pressure of primary loop	2.5MPa	Maximum sea water temperature	105℃
Water temperature at the outlet of the core	213℃	Water productivity	$1.2 \times 10^5 m^3$/day.

(2) Reactor with once through steam generator in the RPV.

In this variant core construction and reactor power are the same as for the Demonstration Heating Plant. That means the reactor power is 200MW. Steam content is about 1%. Pressure of the coolant increases to 4.1MPa, and water temperature at the outlet of the core is 252 ℃. In the RPV, Once Through Steamgenerator (OTSG) instead of the main heat exchangers for the Heating Reactor.

Steam produced from the steamgenerator can be used for propulsion or electrogeneration, It also can be used for sea water desalination.

When the steam produced from OTSG is used for sea water desalination. It goes to a desalination plant steam generator. Where the secondary steam (pressure of the secondary steam is P=1.72MPa and temperature is 205 ℃) is produced and used for desalination. Its condensate goes back to the plant steamgenerator with temperature 120℃. In this case the intermediate loop is working as a separating loop.

When the reactor is working as a power plant, the saturated steam with pressure P=1.9MPa goes to the turbine. At the full power, the electrogeneration efficiency is about 28%. The maximum power is 55MW.

Main parameters of this solution are shown in the Tab. 3.

TABLE 3 MAIN PARAMETERS OF THE ONCE THROUGH STEAM GENERATOR PLANT

Reactor power	200MWt	Steam content of the core outlet	~ 1%
Elect. power (max)	55MWe	Water temperature at the core outlet	135℃
Pressure of the plant steam	1.72MPa	Steam pressure before turbine	1.9MPa
Temperature of the plant steam	205℃	Steam temperature before turbine	210℃
Steam productivity	275T/h	Feed water temperature for the plant SG	120℃
Return water temperature	80℃	power Efficiency	0.28
Water temperature of the core outlet	252℃		

(3) Cogeneration plant with a boiling water reactor.

In this variant boiling is introduced in the 200MW demonstration Heating Reactor design.

Introduction of boiling instead of the once through SG simplified construction. and gave the maximum power efficiency at the lowest pressure.

In the design the pressure increases to 4.1 MPa, steam temperature is 252℃. Power efficiency reached 32%. The water temperature at the core inlet is 204℃. At the same time part of reactor water is parallelly leaded to a steam generator. Secondary steam (pressure 1.72MPa, temperature 205℃) can be used for sea water desalination.

Main parameters of the plant are shown in Tab.4:

TABLE 4 MAIN PARAMETERS OF THE COGENERATION PLANTS WITH A BWR AND A SL

Items	Pl. with a BWR	Pl. with a SL
Reactor power	200MWt	200MWt
Elector- power	62MWe	47MWe
Elector- Generation efficiency	0.32	0.24
Reactor steam pressure	4.1MPa	4.1MPa
Reactor steam temperature	252	252
Water temperature at the core inlet	204℃	204℃
Heating plant steam pressure/temperature	1.72MPa/205℃	
SG pressure	1.72MPa	1.42MPa
Separating loop pressure		4.2MPa
Separating loop temperature		247/195℃

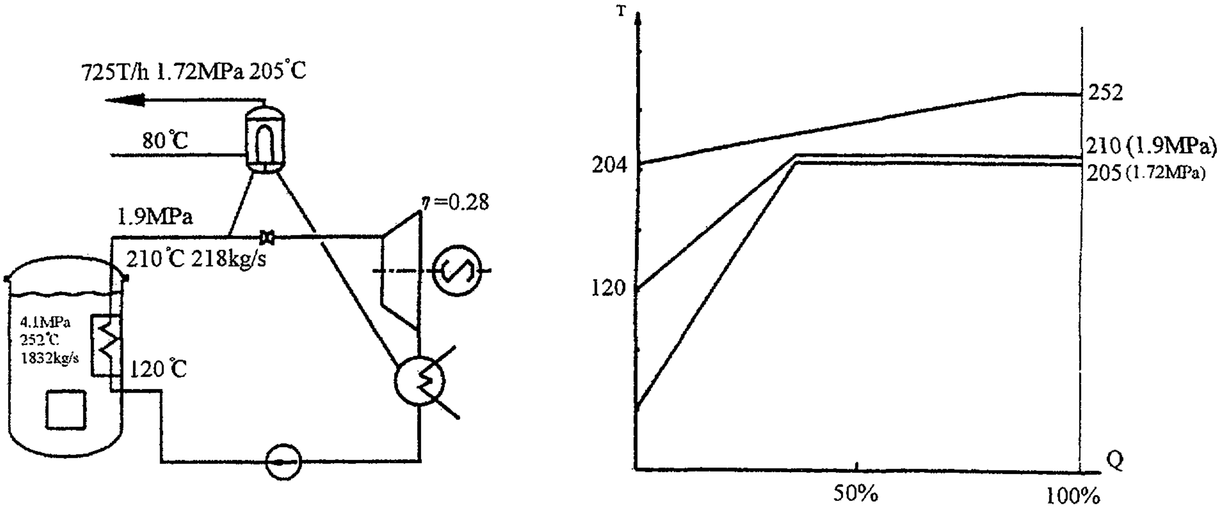


Fig. 4 Cogeneration Plant with OTSG

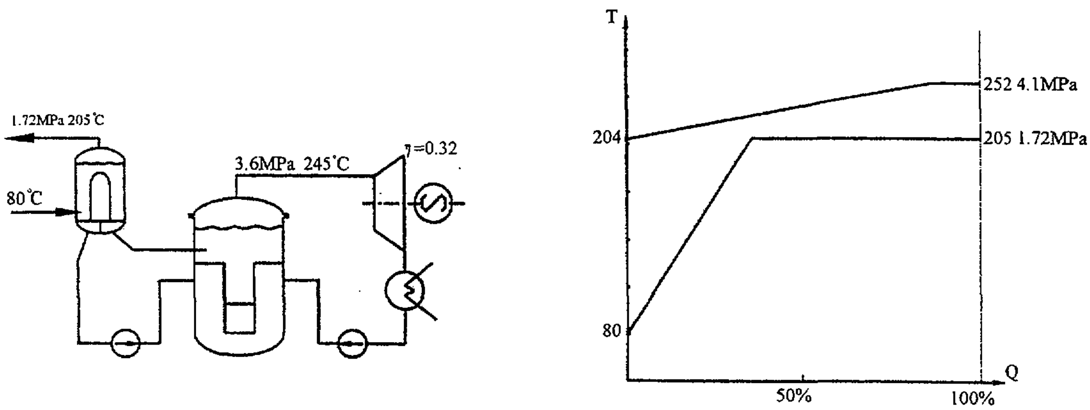


Fig. 5 Cogeneration Plant with BWR

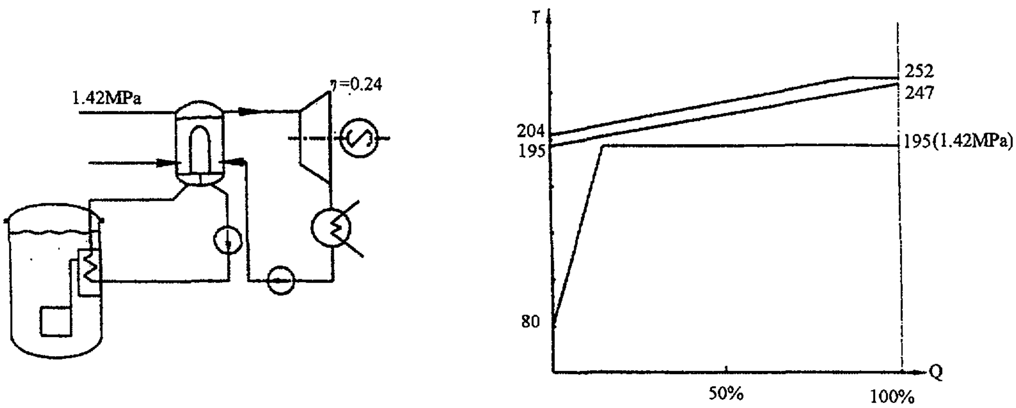


Fig. 6 Cogeneration Plant with a Intermediate Loop

(4) Cogeneration plant with a Separating loop.

In order to effectively separate the power loops from reactor radioactive primary loop,. as in the Demonstration Heating Plant, a intermediate loop solution is adopted in this design variant.

In this variant, reactor work pressure increased to 4.1 MPa. Steam content at the core outlet is about 1%, water temperature 252℃. At the core inlet water temperature is 204℃.

Secondary loop is a separating loop, its pressure 4.2MPa, consisted of main heat exchanger, steam generator (SG) and circulating pumps. In the primary side of the SG temperature are 247/195 ℃, in the secondary side the steam pressure is 1.42MPa, temperature 195 ℃ . Part of steam is used for power generation. The other part of steam is used for desalination or heating. Quantity of heat and power can be regulated by a distribution valve.

If all steam is used for power generation, the maximum power is 47MW and elector-generation efficiency is about 24%.

This variant is more close to the existed Heating Reactor design. Changes are very small. Radioactive water leakage is almost impossible and power productivity is enough high.

Main parameters of the cogeneration Plant with a Separation Loop (SL) are shown is Tab. 4.

4. COMPARISON AND CONCLUSION

In above variants, the first one under the same pressure (≐2.5MPa) as for the existed Heating Reactor design, the reactor can supply some quantity power, but the quantity is small and power efficiency is low (see variant (1)).

If pressure is increased to 4.1MPa, and boiling is introduced in the reactor core, the power productivity will be reasonable increased. In these variants the variant with separating loop more close to original Heating Reactor design. It will be easier to go through the license procedure. But it power efficiency only about 24%.

Once Through Steam Generator variant gives a higher efficiency of but the height of the reactor will be increased, because the steam generator is longer than original heat exchanger.

Boiling water reactor variant gives highest efficiency. But introduce of boiling with a lower pressure requires some additional R & D work. The construction of the core has to do some changes.

At the time being we are working for the variant with a separating loop.

BASIC SAFETY PRINCIPLES OF KLT-40C REACTOR PLANTS

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Abstract

The KLT-40 NSSS has been developed for a floating power block of a nuclear heat and power station on the basis of ice-breaker-type NSSS with application of shipbuilding technologies. Basic reactor plant components are pressured water reactor, once-through coil-type steam generator, primary coolant pump, emergency protection rod drive mechanisms of compensate group-electromechanical type. Basic RP components are incorporated in a compact steam generating block which is arranged within metal-water shielding tank's caissons. Domestic regulatory documents on safety were used for the NSSS design. IAEA recommendations were also taken into account. Implementation of basic safety principles adopted presently for nuclear power allowed application of the KLT-40C plant for a floating power unit of a nuclear co-generation station.

1. KLT-40C REACTOR PLANT FOR FLOATING POWER UNIT OF NUCLEAR HEAT AND POWER STATION

Two-circuit reactor plant (RP) with a vessel-type pressurized water reactor is used for floating power unit of nuclear head and power station. Basic RP components: reactor, steam generators and primary coolant pumps are incorporated by pressure nozzles in a compact steam-generating block. KLT-40C RP characteristics are given in Table 1.

Creation of KLT-40C RP is carried out on the basis of existing ice-breaker-type RP and established shipbuilding technologies. Technical solutions are validated by 39-five year experience of operation under navigation conditions in Arctic seas. Achieved service life of the ship reactor plant equipment and systems exceeds 130 000hrs. Basic equipment of KLT-40C RP represents industrially fabricated equipment used for serial KLT-40-type reactor plants. The reactor plant meets the quality assurance requirements according to international QA code ISO-9000. Pressurized water reactor (Fig. 1) with forced coolant circulation in primary circuit is used. The reactor is composed pressure of vessel, cover, internals removable block, core, drive mechanisms for CG and EP. The vessel is forged-welded structure. Vessel structural material is heat-resistant high-strength perlytic steel with anticorrosive overlaying.

TABLE I. KLT-40C RP NOMINAL CHARACTERISTICS

Characteristic	Value
Thermal power, MW	150
Steam output t/h	240
Pressure of steam after SG, MPa	3,8
Temperature of super heated steam, °C	290
Temperature of feed water, °C	170
Service life, years	35-40
Time between core refueling, years	2, 5-3

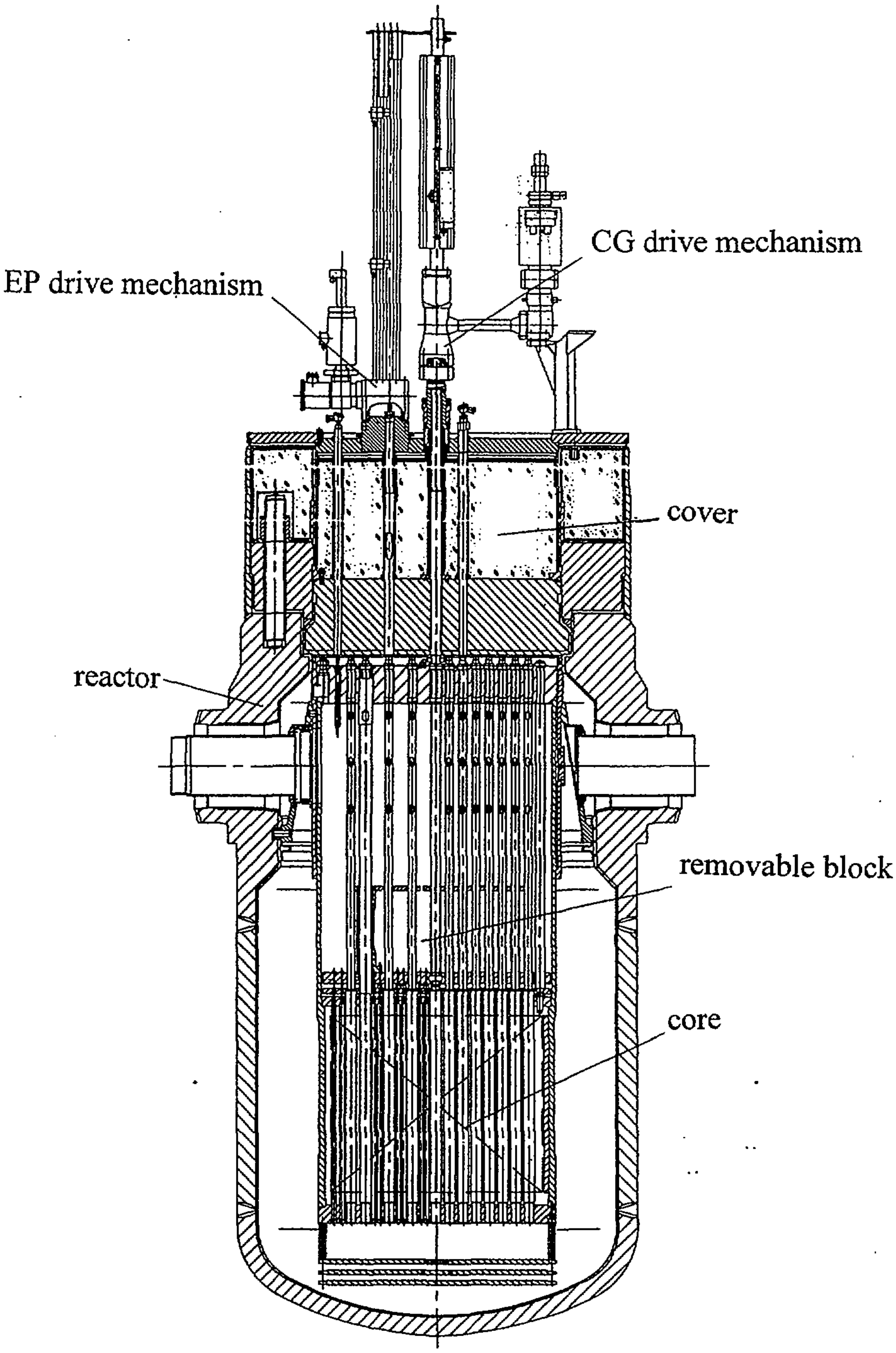

Fig.1. Reactor

The core has heterogeneous arrangement and uses dispersion-type nuclear fuel. Core consists of a set of FA and sets of reactivity control and safety rods. FAs incorporate burnable poison (gadolinium) rods to compensate the core excessive reactivity. The core uses smooth-pin type fuel element with a cladding made of zirconium alloy.

Fuel composition is selected taking into account of high compatibility with zirconium-alloy cladding, structural stability at high temperature, sufficient corrosion-erosion resistance in water environment, high heat conductivity. The core possesses negative power and temperature reactivity coefficients within the entire range of operating parameters variation during core life period.

SG is once-through coil-type heat exchanger with steam generation within tubes (Fig. 2). The SG's tube system consists of cylindrical helical coils is made of titanium alloy. The SG's shell structural material is low-alloyed steel with anticorrosive overlaying. The PCP is canned centrifugal one-stage pump with screened dual-speed (two-winding) asynchronous motor. Capacity - 870m3/h, head -0,38Mpa. The pump casing items are fabricated of austenitic stainless steel. Electric motor's rotor is fabricated of ferritic stainless steel. The pump bearings are lubricated and cooled, as well as both electric drive rotor and stator are cooled by primary coolant which circulates in an autonomous loop. Heat from the loop is removed by cooling water.

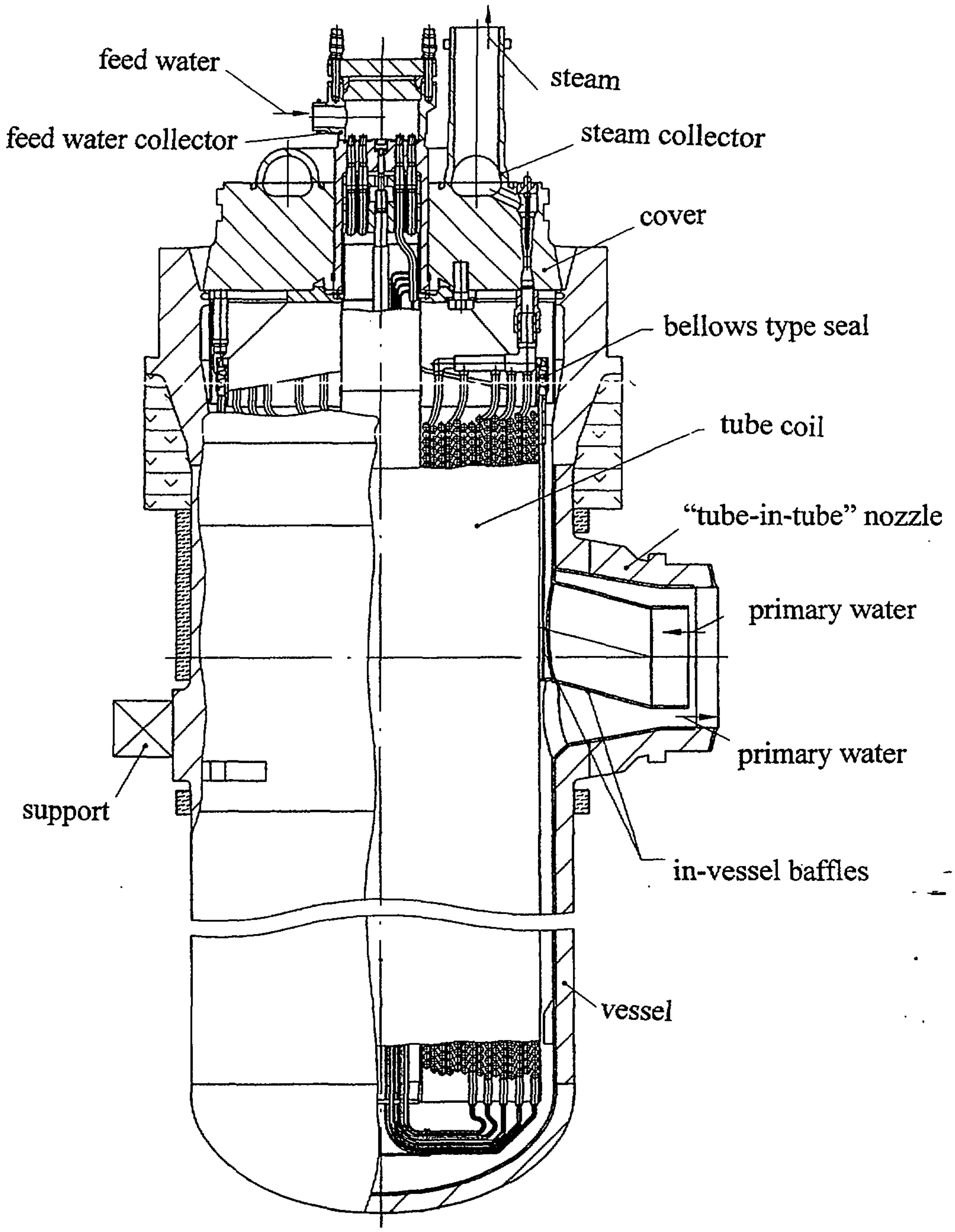

Fig.2. Steam generator

The emergency protection rod drive mechanism is an electromechanical type actuator consisting of rack mechanism with a spring, asynchronous motor and electromagnet. Scram time is not more than 0,5s. Absorber rods are inserted under spring action when holding electromagnets are de-energizated.

The drive mechanism of compensate group consists of, screwed reducer, stepped electric motor, movement sensor, reference points sensors. Working speed is 2mm/s. Emergency speed of lowering by electric motor on trip signals is 4mm/s. Average speed of lowering under gravity is 30 to 130mm/s.

The steam-generating block is arranged within metal-water shielding tank's caissons (Fig. 3). Auxiliary equipment items including pressurizes, filter, heat exchangers are also located in the tank's caissons. All radiation sources are surrounded by shielding. RP shielding consists of the tank, removable of dry shielding blocks and peripheral shielding on protective enclosure walls.

Main codes are used:

- "General Principles of Safety Provision for NPPS" (OPB-88).
- "Rules of Nuclear Safety for Nuclear Reactor Plants of NPPS" (PBYA RU AS-89).
- Norms of Radiological Safety NRB-96. Hygiene's standards GN 2.6.1.054-96 Law of Russian Federation "Radiological safety of population".
- Rules of nuclear ships classification and building for Russia's sea ship navigation register.

Other effective guides and rules for nuclear stations. IAEA recommendations are taken into account at KLT-40 RP development:

- basic principles of safety referring to safety control and full-scope realization of in-depth protection strategy;
- main technical principles including the use of approved engineering-technical practice, assuring quality, account of man-factor, systemic analysis, evaluation and substantiation of safety using deterministic and probabilistic methods, analysis, account of analogues operation experience;
- execution of measures on radiation protection of personnel, population, on environment under normal operation and at emergencies including postulated severe accident with core damage;
- recommendations on specific engineering decisions referring to all-sided development of intrinsic safety, use of safe failure principle, redundancy, diversity, spatial separation of means and systems important to safety, use of passive systems and self-acting safety devices, assurance of accident control capabilities.
- recommendations on specific engineering decisions referring to all-sided development of intrinsic safety, use of safe failure principle, redundancy, diversity, spatial separation of means and systems important to safety, use of passive systems and self-acting safety devices, assurance of accident control capabilities.

2. MAIN SAFETY PRINCIPLES

Next main safety principles are taken into account at KLT-40 RP:

- use of pressurized water reactor with developed intrinsic safety provided by core feed back, reactor thermal inertia, coolant natural circulation in emergency situations, etc.

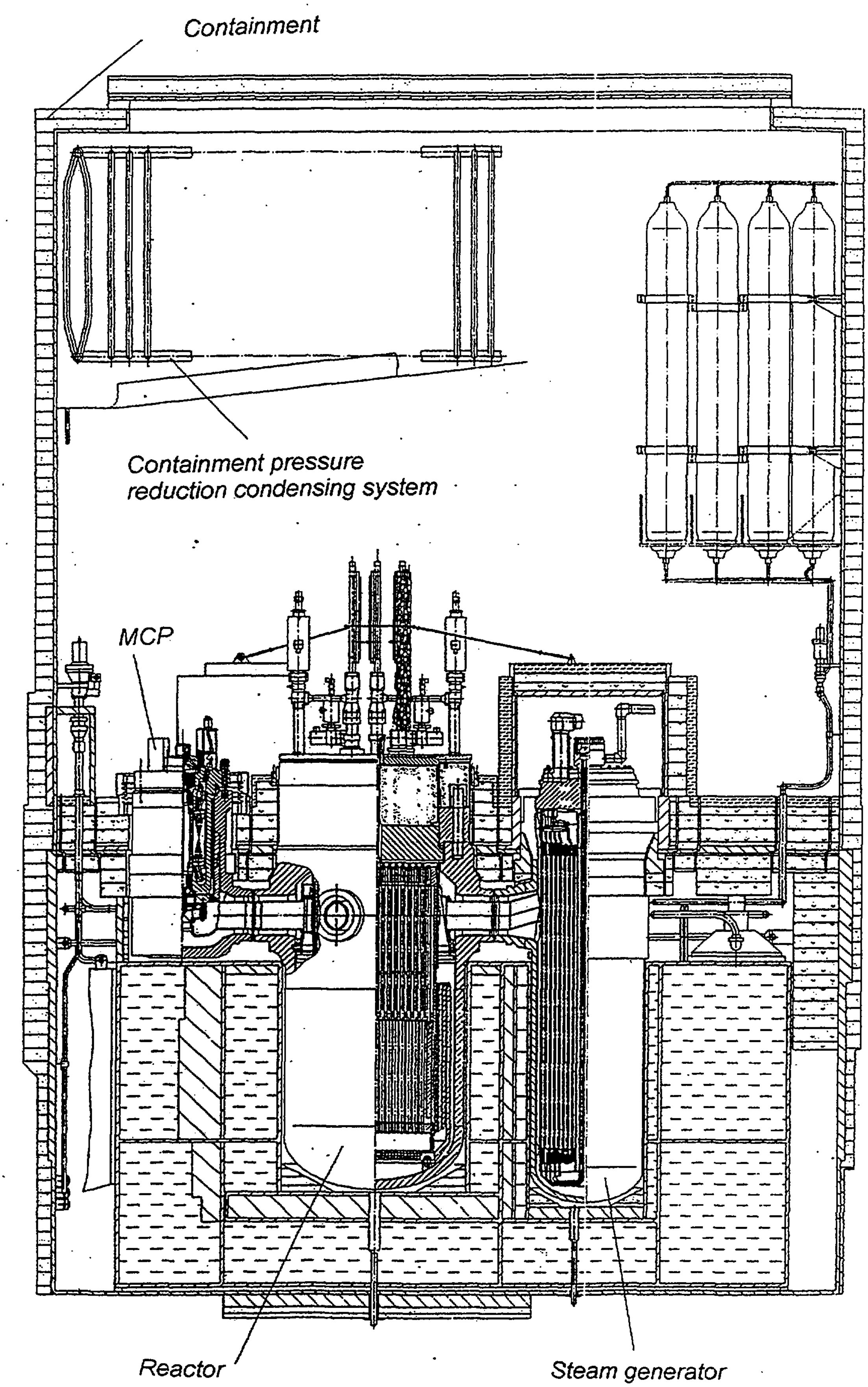

Fig.3. Reactor plant arrangement

- Provision of in-depth functional and physical protection by effective safety barriers and isolation systems eliminating the discharge of radioactive products beyond FPU boundaries at most severe accidents accompanied by additional failures (Fig. 4).
- establishment of protection against inner and outer impacts.
- Use of conservative approach when developing physical barriers, safety systems, choice and substantiation of initial events of accidents and their development scenarios.
- Use of physically separated safety systems of both active and principle of operation without use of external power sources and without personnel intervention.
- Use of highly reliable self-diagnosing automatic control systems and operator logical infonnation support systems.
- Use of diagnostic control systems allowing to determine the actual state and remaining life of the nuclear plant most responsible equipment and pipelines.
- Realization of maximally attainable reactor plant waste-free technology limiting formation of liquid radioactive waste during the reactor plant operation.

Primary circuit system (Fig. 5) includes:

- main circulation circuit;
- pressurization system;
- purification and cooldown system,

Next safety systems are used at KLT-40 RP:

- reactor trip system;
- emergency cooldown system;
- emergency core cooling system;
- containment emergency pressure reduction system; containment system;
- reactor caisson water filling system.

Reactor trip system includes;

- four independent groups of safety rods with individual drive mechanisms actuated passively under compressed springs action;
- five independent compensating groups (CG) with individual drive mechanisms ensure reactor trip on the control system signals by:
- lowering by electric drive with emergency speed; lowering under gravity; liquid absorber injection system (back up system).

It is provided for lowering under gravity using safety devices for CG-drives de-energization executing by increase of primary circuit pressure.

Emergency cooldown system consists of':

- active cooldown train through primary circuit purification system's heat exchanger with heat transfer via third circuit to ambient sea water;
- active cooldown train through steam generators with heat removal to process condenser and then to sea water;
- two passive cooldown trains through steam generators with heat removal to emergency cooldown water tank heat exchanger and then to the atmosphere by means of water evaporation in tank.

The system is actuated both control system and direct action of process parameter, i.e. by increased primary pressure using hydraulically- controlled air distributors.

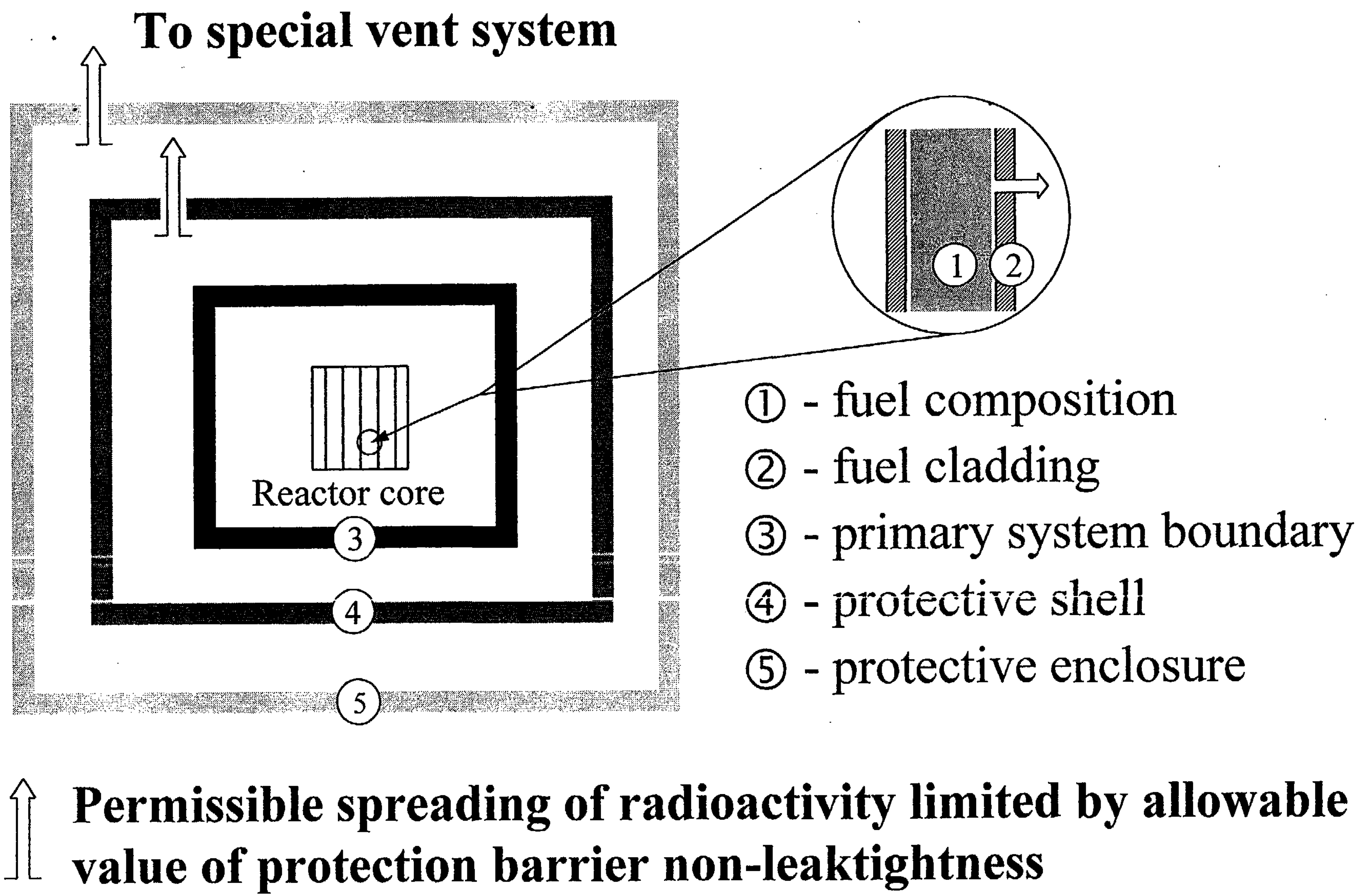

Fig.4. Schematic diagram of protective phisical barriers

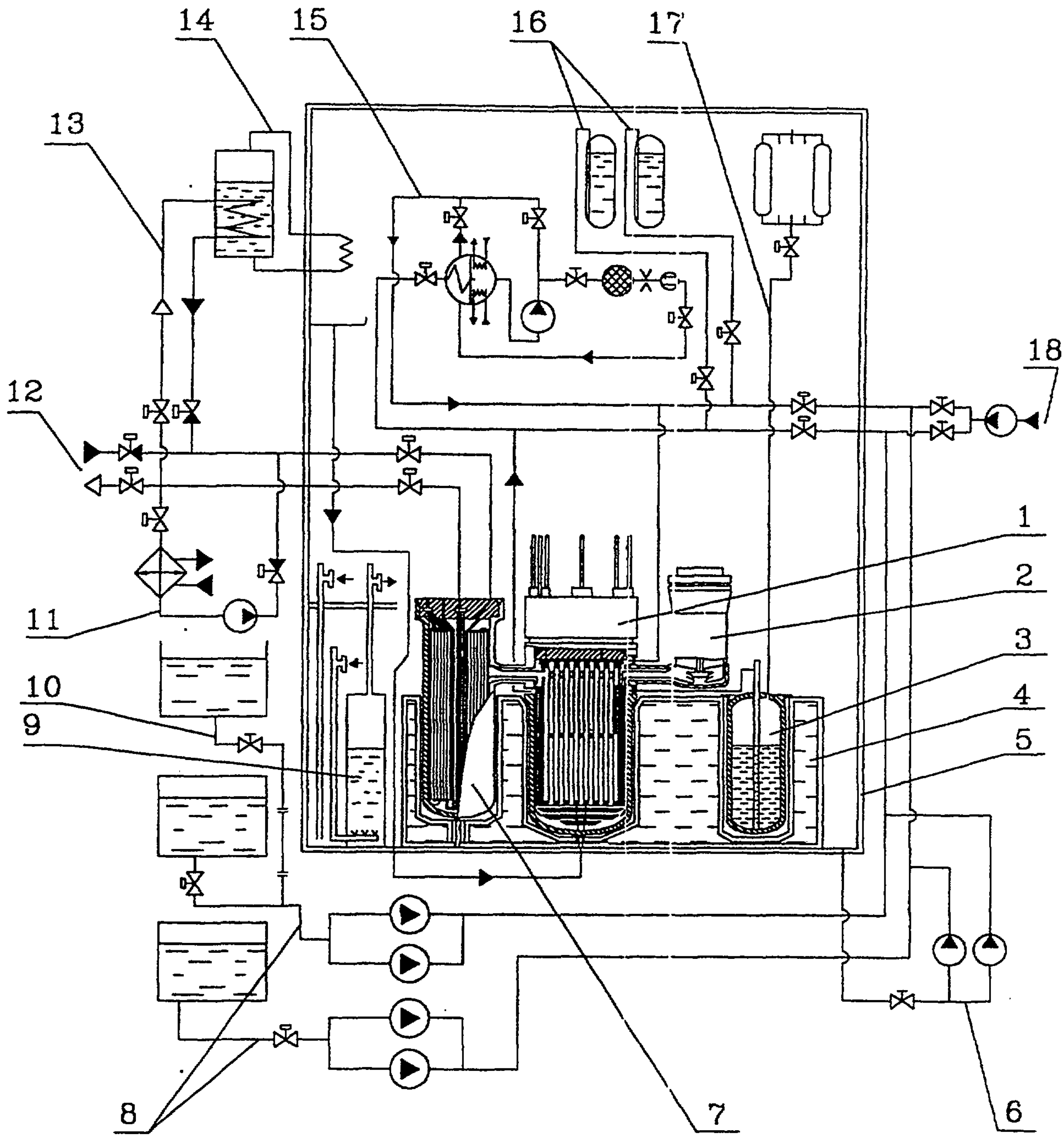

Fig.5. Reactor plant principal flow diagram

1 - reactor; 2 - reactor coolant electric pump; 3 - pressurizer; 4 - metal-water shielding tank; 5 - protective shell; 6 - recirculation system; 7 - steam generator: 8 - engineered emergency core cooling system; 9 - bubbler system for pressure supressure in protective shell; 10 - liguid absorber removal system; 11 - engineered emergency shutdoun cooling system; 12-to steam turbine plant; 13 - passive emergency shutdoun cooling system; 14 - protective shell pressure supressure emergency condensation system; 15 - primary circuit purification and cooldown system; 16 - passive emergency core cooling system; 17 - pressurization system; 18 - from STP

Water inventory in the tank ensures the RP maintaining in safe state for not less than one day without personnel intervention.

Emergency core cooling system consists of two trains. Each train meets the single failure principle. System includes high and low pressure sub-system. High-pressure sub-system includes passive (hydro-accumulators) and active pumps and water storage tanks) features for water injection in reactor. Low-pressure sub-system ensures returning a condense accumulated in containment, into the reactor by recirculation pumps.

Containment emergency pressure reduction system uses passive principle of operation. Suppression of emergency pressure is provided by steam condensation in bubbling tank, on heat exchange's surfaces and on containment walls. Reactor caisson water filling system is intended to protect reactor vessel against melting through during severe beyond design-basis accident involving loss-of-coolant and core and melt. Water and condensate are supplied to the reactor caisson from bubbling tank, condensate collector and shielding tank's plating (elevation head). System works passively by steam condensation in pressure reduction system's heat exchangers and by condensate supply into reactor caisson under gravity. Containment a robust leak-tight compartment designed for internal pressure of 0,5MPa (abs.). There no are systems and equipment within the containment containing sea water. A partial vacuum of 300 Pa is maintained in the containment during the RP power operation.

Special system is provided for filling the containment with water and its subsequent sealing to exclude destruction of containment at FPU sink. Main trends of severe accident control strategy limitation of core damage size:

- prevention of core melting;
- preservation of core vessel integrity with confinement of core materials inside the vessel; preservation of protection envelope integrity with account of a severe accident consequences; limitation of escape of radioactive products into the environment.
- Reactor intrinsic safety is due to:
- negative coefficients of core reactivity providing for self-termination of chain reaction in the core at unauthorized rise of reactor power and temperature;
- passive principle of reactor shutdown by working members;
- high accumulating capability;
- elimination of large diameter pipelines in the primary circuit-limiting outflow of coolant at the circuit depressurization.

TABLE II. KLT-40C RP DOSES LIMITS AND CRITERIA

Effective irradiation dose	Dose limits for personnel	Dose limits for population
At normal operation	2 rem/year	0,1 rem/year
At design basis accidents	10 rem	0,5 rem (on SPZ boundary)
At beyond design basis accidents	20 rem	0,5 rem (on SPZ boundary)
Limits of damageability expressed through primary circuit fragmentary activity	Operation limit Safe operation limit	$10^{-3} Ci/kg$ $10^{-3} Ci/kg$

3 ENVIRONMENTAL SAFETY OF ATETS WITH KLT-40C RP

Based on operation experience of nuclear ice-breaker fleet (more than 150 reactor-years nether):

☑ no single event personnel overexposure was registered;

☑ average close is in 10 times lower than allowable limit.

Environmental safety of atest with KLT-40C RP is shown in table III.

TABLE III. ENVIRONMENTAL SAFETY OF ATETS WITH KLT-40C RP

Mode	Name	Value
Normal Operation	Radioactivity discharge	
	Σ IRG	10 Ci
	Σ I and Cs isotopes	<0, 01 Ci
	Irradiation dose for population	~0, 01 mrem
	Heat discharge	
	To atmosphere	270 кW
	To outboard water	1650 кW
Maximum design basis accident	Radioactivity discharge	
	Σ IRG	11 Ci
	Σ I isotopes	0,01 Ci
	Σ Cs isotopes	~10^{-6} Ci
	Irradiation dose for population	~0, 02 mrem

4. RADIOACTIVE WASTES (RAW) MANAGEMENT PRINCIPLES

All radioactive wastes are stored on FPU board during operation.

Wastes are not neither stored, processed nor disposed FPU sitting water area.

Tanks and casks arranged in shielded compartments are used to collect RAW and store them at FPU.

Gaseous RAW are practically absent during operation including refueling operations.

Amount of RAW is mainly determined by refueling operations and amounts to (for one reactor plant annually):

☑ liquid RAW is 8 m^3;

☑ solid RAW is 2.5 m^3.

Liquid RAW are as a rule low active with average activity of 10^{-6} Ci/kg. More than 70% solid RAW are low active as well.

5. ENVIRONMENTAL AND ECONOMIC INDEXES OF NPPS WITH KLT-40 RP

NPP with KLT-40C RP

☑ Saves 300000t of equivalent fuel per year.

☑ Saves 400 mln m^3 of air oxygen per year.

☑ Does not charge the atmosphere with tons per year of:

	compared to coal-fired plant	compared to oil-fired plant *
solid particles (dust, ashes)	730	130
sulphuric acid	12000	8400
nitrogen oxides	3400	2200
vanadium pentoxide	4	56

*) The data are given for a 300 MW(th) thermal stations sulphur content in coal and mazut is 2% efficiency of solid particles separation is 99%.

The KLT-40C NSSS created with reliance upon the proven technology of nuclear propulsion plants of icebreaker type, involving also utilization of ship-building technologies, is a reliable and safe plant capable of minimizing environmental impacts compared to alternative fossil-fired power plants. NPP with such NSSS can be situated in close vicinity of settlements.

A FLOATING DESALINATION/CO-GENERATION SYSTEM USING THE KLT-40 REACTOR AND CANADIAN RO DESALINATION TECHNOLOGY

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Abstract

As the global consumption of water increases with growing populations and rising levels of industrialization, major new sources of potable water production must be developed. To address this issue efficiently and economically, a new approach has been developed in Canada for the integration of reverse osmosis (RO) desalination systems with nuclear reactors as an energy source. The resulting nuclear desalination/cogeneration plant makes use of waste heat from the electrical generation process to preheat the RO feedwater, advanced feedwater pre-treatment and sophisticated system design integration and optimization techniques. These innovations have led to improved water production efficiency, lower water production costs and reduced environmental impact.

The Russian Federation is developing the KLT-40 reactor for application as a Floating Power Unit (FPU). The reactor is ideally suited for such purposes, having had many years of successful operation as a marine propulsion reactor aboard floating nuclear powered icebreakers and other nuclear propelled vessels. Under the terms of a cooperation agreement with the Russian Federation Ministry of Atomic Energy, CANDESAL Enterprises Ltd. has evaluated the FPU, containing two KLT-40 reactors, as a source of electrical energy and waste heat for RO desalination. A design concept for a floating nuclear desalination complex consisting of the FPU and a barge mounted RO desalination unit has been analyzed to establish preliminary performance characteristics for the complex.

The FPU, operating as a barge mounted electrical generating station, provides electricity to the desalination barge. In addition, the condenser cooling water from the FPU is used as a source of preheated feedwater for the RO system on the desalination barge. The waste heat produced by the electrical generating process is sufficient to provide RO feedwater at a temperature of about 10°C above ambient seawater temperature. Preliminary design studies have indicated that under these conditions approximately 100,000 m^3/d of potable water can be produced. The use of preheated feedwater results in an improvement in water production efficiency of up to about 15% relative to a system operating at the ambient seawater temperature.

This preliminary design study has shown that significant improvements in the cost of water production can be achieved through this "marriage" of Russian small reactor technology and Canadian RO technology. The potential benefits warrant further detailed evaluation followed by a demonstration project.

A FLOATING DESALINATION/COGENERATION SYSTEM USING THE KLT-40 REACTOR AND CANADIAN RO DESALINATION TECHNOLOGY

1. INTRODUCTION

In many regions of the world the supply of renewable water resources is inadequate to meet current needs, and that from non-renewable sources is being rapidly depleted. Since the worldwide demand for potable water is steadily growing, the result is water shortages that are already reaching serious proportions in many regions, with the threat of global water starvation continuing to grow. To mitigate the stress being placed on water resources, additional fresh water production capability must be developed. For many regions seawater desalination is the best alternative. The main drawback of desalination, however, is that it is an energy intensive process. Therefore, the increasing global demand for desalted water creates a tremendous collateral demand for new sources of electrical power. Since water is an undeniable life sustaining resource, improvements in the efficiency of energy utilization must be considered a significant benefit to both the environment and the consumer. CANDESAL Enterprises Ltd. is a Canadian company working internationally to improve the energy efficiency and economics of fresh water production and to deliver that technology to markets where such facilities are most required.

In December 1994 the International Atomic Energy Agency (IAEA) sponsored one of a series of conferences on nuclear desalination in Cairo, Egypt. The Russian delegation at that conference met with the CANDESAL delegation to discuss a project that was in the planning stages by the Ministry of Atomic Energy's (MINATOM) Design and Engineering Bureau, OKBM. The project was to design and build a series of floating barges using nuclear reactors for electrical generation and as an energy source for desalination. The initial objective was to meet the requirements of the resource industries working in Russia's northern littoral, and then to use these floating desalination/cogeneration barges to help meet the world demand for additional energy and water production, particularly in the Middle and Far East. In the conceptual stages of the project, primarily distillation systems were being considered. However, the work done by CANDESAL was clearly showing the potential for significantly reducing the production costs of potable water through the use of reverse osmosis (RO) technology. In view of the potential benefits, it was agreed that a cooperative venture should be undertaken to evaluate the use of RO desalination with the Russian power barges. Following these preliminary discussions a Memorandum of Understanding (MOU) between MINATOM and CANDESAL was signed in 1995 for a period of two years. This MOU has recently been extended for a period of five years.

2. WATER SCARCITY, A GROWING GLOBAL CONCERN

There is a growing international recognition that the shortage of adequate supplies of potable water, which has already reached critical proportions in many areas of the world, is one of the major problems facing society as the 21st century approaches. There is also an acknowledged need for additional electrical generation capacity throughout the developing world. Frequently, these shortages in potable water supply and electrical energy exist together.

The supply of naturally occurring fresh water available for human use is limited. It consists of non-renewable sources such as aquifers and other reservoirs that are not recharged as they are used, and renewable sources such as lakes, rivers, reservoirs and other sources that are replenished by the annual water, or hydrologic, cycle. However, the amount of water available in a given location as a result of the natural water cycle is essentially fixed. Thus as the population increases, the annual water supply per person, which is a general indicator of water security, decreases. The per capita water supplies worldwide are approximately one third less now than in 1970 due to population growth since that time.[1] The water supply crises which already exist and are projected over the next few decades have received much attention recently.[1-6] Population Action International expressed the concern quite succinctly when they stated[6] that "without water, economic development becomes virtually impossible and conflict over scarce resources virtually inevitable" and "availability of and access to clean water and sanitation are among the most important determinants of the health of individual human beings."

Figure 1[7] illustrates the percentage of urban populations with access to safe drinking water. It is estimated that around the world approximately one billion people do not have access to safe water, and 1.8 billion do not have adequate sanitary facilities. The lack of water is a detriment to human health to these people.

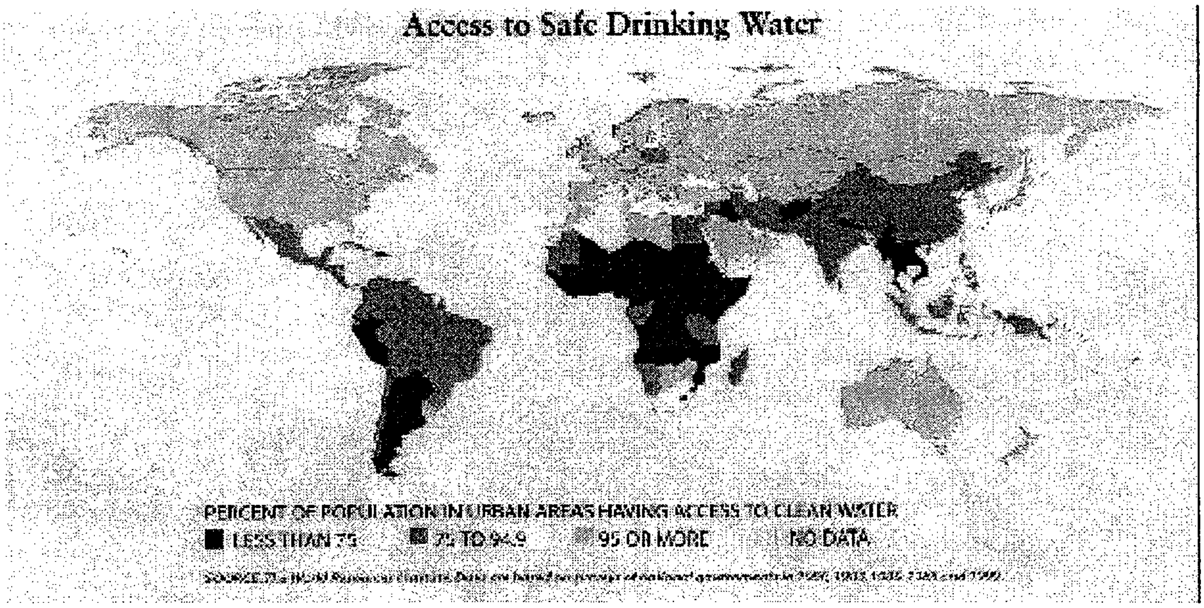


Figure 1
Percentage of Urban Population Having Access to Safe Drinking Water

Hydrologists have adopted the concept of a "water stress index" based on an approximate minimum level of water required per capita to maintain an adequate quality of life in a moderately developed country in an arid zone. Although the indicators are only approximate, it has been found that a country or region whose renewable fresh water availability exceeds about 1700-2000 m^3 per person per year will suffer only occasional or local water problems. Below this threshold the lack of water begins to become a serious problem. Two rough benchmark levels have been adopted:

- **Water Stressed:** A country or region is considered **water stressed** if the availability of renewable fresh water supplies falls between 1000 and 2000 m^3 per person per year.
- **Water Scarce:** A country or region is considered **water scarce** if the availability of renewable fresh water supplies is less than 1000 m^3 per person per year. At this level, the chronic lack of water begins to hamper human health and well being, and is a severe restraint on food production, economic development and protection of natural systems. Below 500 m^3 per person per year, there is considered to be an **absolute scarcity** of fresh water.

Since renewable fresh water resources are essentially constant, the per capita availability falls as population rises, pushing more and more countries over time into water stress and water scarcity. Figure 2 illustrates on a global basis the countries projected to experience water stress or water scarcity by the year 2025, based on a United Nations medium population projection.[6]

3. Desalination As A Source Of Potable Water

Seawater is the largest source of available water. It is estimated that only 2.5% of the earth's 1.4 billion km^3 of water is fresh water, and 69% of that is in the form of ice (polar ice caps and glaciers) or in underground aquifers too deep to tap. The amount of fresh water that is available is essentially fixed, and as the population continues to grow, the per capita availability shrinks. Seawater, on the other hand, is available in essentially unlimited quantities in the foreseeable future, and is still relatively unpolluted compared with natural fresh water sources many regions of the world.

As other sources of water have become increasingly scarce and increasingly expensive to make available, desalination of seawater has become a more attractive option for the production of potable water. By the late 1960s commercial units of up to 8,000 m^3/d were beginning to be installed in various parts of the world. These were primarily thermal distillation units, but by the 1970s commercial membrane processes were also becoming available. Figure 3 illustrates the rapid growth

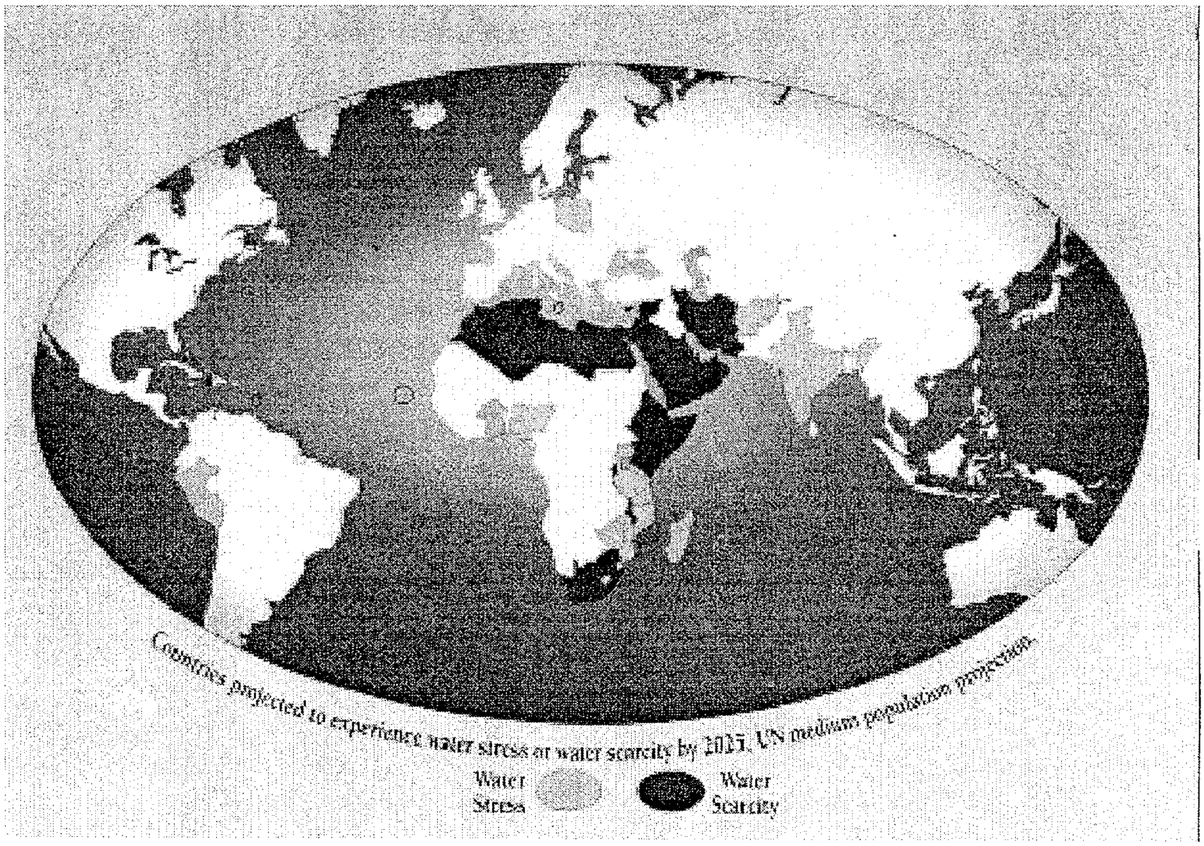


Figure 2
Population and Water Availability

of desalination as a source of potable water since that time. As of 1995, there were about 20 million m^3/d of desalination capacity either installed or contracted for around the world. That number is expected[8,9] to increase to well over 35 million m^3/d over the next ten years.

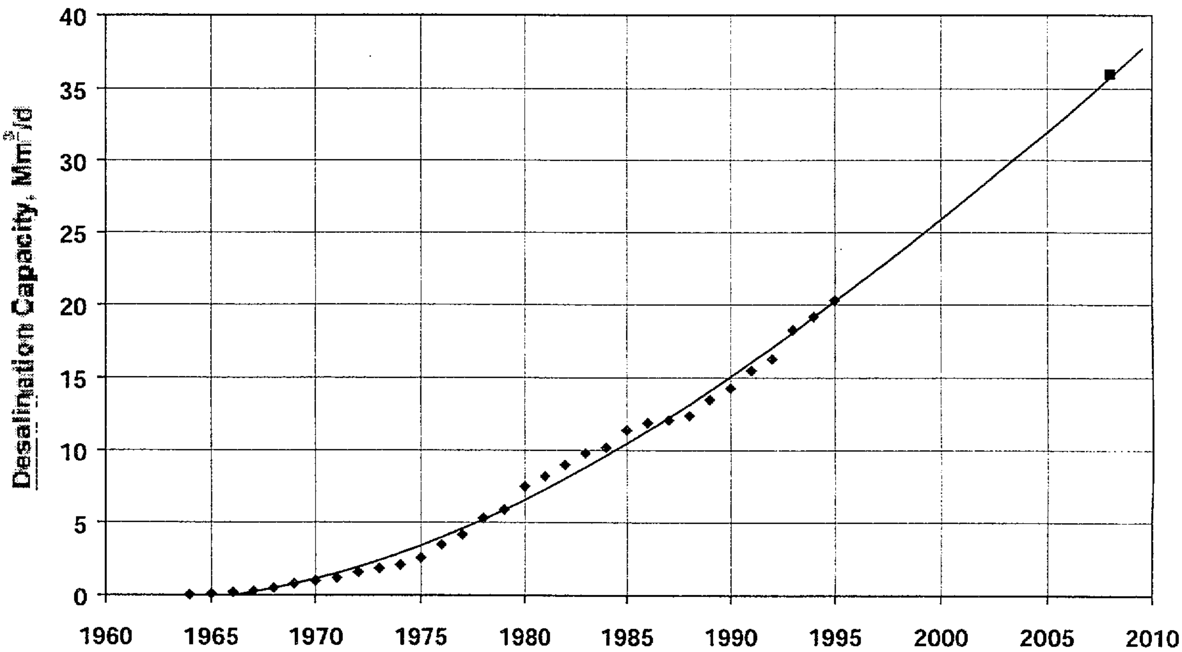


Figure 3
Worldwide Contracted Cumulative Seawater Desalination Capacity

Recognizing the severity of the problem of global water shortages and the immensity of the energy requirement necessary to adequately address the problem, the International Atomic Energy Agency (IAEA) has been working since 1989 to assess the technical and economic potential for using nuclear reactors as a source of energy for seawater desalination. The results of studies[8-10] carried out by the IAEA and its Member States have shown that the use of nuclear energy for desalination is technically feasible and in general economically competitive with fossil-fueled energy sources. As a result of the success of its efforts to date, the IAEA is expanding its nuclear desalination program, and is actively supporting efforts amongst Member States to introduce nuclear desalination demonstration projects.

4. A FLOATING NUCLEAR DESALINATION/COGENERATION SYSTEM

4.1 Development of the CANDESAL Design Approach

Because of the pressing need for additional large scale water production capability, the focus of CANDESAL's early design concept development work was placed first on the use of the CANDU nuclear reactor as an energy source for desalination. Two approaches were considered in preliminary studies: the use of electrical energy for reverse osmosis (RO) and the use of process steam from the nuclear steam supply system to provide the energy for a multi-effect distillation system. This latter approach was found to require changes to the balance of plant design that were both expensive to implement and led to reduced electrical generating efficiency to such a degree that the total water and electrical production capacity was not as great as that which could be achieved using RO.

Having selected the RO process, it was then recognized that improvements in the efficiency of energy utilization could be achieved by taking advantage of waste heat normally discharged from the reactor through the condenser cooling system. Use of the condenser cooling water as preheated feedwater to the RO system improves the efficiency of the RO process, and therefore the economics of water production. As the development work progressed, it was also found that further improvements could be achieved by taking a systems approach to optimizing the design. Hence a strong emphasis has been placed on the integration of the energy and water production systems into a single, optimized design for the cogeneration of both water and electricity. Details of the CANDESAL approach to the application of RO seawater desalination technology have been described elsewhere[11-15].

This approach to the integration of seawater desalination systems with nuclear reactors has the advantage of maximizing the benefits of system integration while at the same time minimizing the impact of physical interaction between the two systems. In essence, the reactor operates without "knowing" that there is a desalination plant associated with it. Transients in the desalination plant do not have a feedback effect on reactor operation. This is extremely important, since there must be a high degree of assurance that unanticipated operating transients in the desalination unit do not have an adverse impact on either reactor safety or operational reliability. Conversely, it would also be undesirable to have reactor shutdowns, whether unanticipated or for planned maintenance, that would require shutdown of the water production plant.

Hence as the CANDESAL nuclear desalination/cogeneration system design has developed, it has evolved in a direction which allows standardized reactor systems to be used without modification, while at the same time accruing significant benefits from the systems integration due to improved performance characteristics and energy utilization. Furthermore, because of the "loose" coupling to the reactor system, the use of RO with preheated feedwater is ideally suited for application with small marine reactors such as the Russian KLT-40, since it allows the economic benefits of integrating the energy generation and desalination systems without sacrificing the flexibility of the smaller reactors, including its use in floating systems.

4.2 The Floating Nuclear Seawater Desalination/Cogeneration Complex

The floating nuclear seawater desalination/cogeneration plant consists of two floating structures: a floating power unit (FPU) and an RO seawater desalination barge. This arrangement, which separates power generation and potable water production, has certain advantages over an arrangement in which they are combined on one floating structure. It allows either of the two systems to be used independently, and it allows manufacture in two separate shipyards if desired. It also simplifies a solution to the problem of preserving efficient potable water production from the complex when the reactors are shutdown by supplying the desalination ship with electric energy from a shore grid. Advantages of a floating desalination/cogeneration complex relative to a shore-based installation include:

- the possibility of plant manufacture and testing at a shipyard in the Supplier country;
- high fabrication quality at a shipyard and "turn-key" delivery in a short period of time;
- convenient maintenance by a floating base at a mooring site and decommissioning by tugging to the Supplier's country;
- commercial production and long-term confirmation of service life characteristics of the KLT-40-type reactor plants and desalination units;
- the possibility of installation in different coastal regions of the world; and
- ease of redeployment of the facility to other locations if the need should arise.

The design and industrial enterprises of Russia are working on the development of the floating nuclear power station for the country's northern regions. This can serve as a prototype for application of the FNPS in a desalination/cogeneration complex.

The second structure, the desalination barge, is a non-self-propelled structure housing systems and equipment providing for the supply of seawater to the desalination system, pretreatment of the feed supply, desalination, supply of desalinated water to the on-shore distribution system, and cleaning of the desalination units.

Based on this two barge concept, OKBM has established design and performance characteristics for the FPU sufficient to allow preliminary design work to proceed on the desalination barge. These characteristics have been provided to CANDESAL for use in the desalination system analysis. The conditions taken as input to the analysis were:

Reactor/Condenser Design Parameters

• Number of reactors per system	2
• Electrical power production, per reactor	35 MWe
• Total electrical production	70 MWe
• Electrical power consumed as "house load"	5 MWe
• Condenser cooling water flow rate, per reactor	5400 m^3/hr
• Total condenser cooling water flow rate	10,800 m^3/hr
• Temperature rise across condenser	10°C

Seawater Conditions

• Seawater temperature (design value)	18°C
• Seawater total dissolved solids (TDS)	38,500 ppm

4.3 RO System Conceptual Design Analysis

The RO system design is based on using spiral wound Dow FilmTec high rejection membranes for seawater desalination. Condenser cooling water being discharged from the reactor's condenser cooling water system at a temperature 10°C above ambient seawater temperature is used as feedwater for the system. The water first passes through ultrafiltration (UF) pretreatment modules. The filtrate from the UF units is sent to capacitance tanks, from which suction is taken for the RO modules. The feedwater is pumped to high pressure (1000 psi) and then passes through the RO modules. The permeate from the RO membranes is discharged temporarily into potable water storage tanks, and from there goes to an off-ship distribution system. Brine concentrate from the RO system is discharged back into the sea. Chemical pretreatment of feedwater and post-treatment of potable water is included as necessary.

The RO system design has been for optimum water production at 28°C, while still maintaining the capability for water production within the design specification of the RO membranes when operating at ambient seawater temperature. Based on the design inputs specified above, the desalination plant, consisting of a UF pre-treatment system and the RO desalination system has the following characteristics:

UF System

• Feed flow into UF modules	259,000 m^3/d
• UF system recovery	90 %
• UF filtrate flow	233,000 m^3/d
• Number of UF vessels	1800
• UF membranes per vessel	4
• Total number of UF membranes	7200

RO System

• Feed flow into RO system (UF filtrate)	233,000 m^3/d
• RO system design temperature	28 °C
• RO system operating pressure	1000 psi
• RO system recovery (at 28 °C)	43.3 %
• Permeate flow	101,000 m^3/d
• Number of RO vessels	1500
• RO membranes per vessel	7
• Total number of RO membranes	10,500

Overall System Characteristics

• Number of 10 vessel by 10 vessel UF arrays	18
• Dimensions of 10x10 UF array (Includes 4m x 5m x 6m space at end of array for pumps, headers, etc.)	4mW x 5mH x 14mL
• Number of 10 vessel by 10 vessel RO arrays	15
• Dimensions of 10x10 RO array (Includes 4m x 5m x 6m space at end of array for pumps, headers, etc.)	4mW x 5mH x 14mL
• Weight of membrane and vessel arrays (Includes water contained in the membranes/vessels)	1000 metric tons
• Total weight of system	15,00-20,00 metric tons
• Electrical power consumption for water production	18-19 MWe

4.4 High Ambient Seawater Temperatures

The above design characteristics are based on 18°C seawater. Calculations have also been carried out for seawater temperatures of 30°C (40°C preheated RO feedwater), which is more representative of temperatures along the coastline of some of the countries where a floating nuclear desalination system would have potential application. Analysis results are very similar, except that only 1200 RO vessels, containing a total of 8400 membranes, would be required. Hence only 12 RO arrays would be required under these conditions. However, since it is important that the design be able to accommodate a range of seawater temperatures, at this preliminary stage the 18°C seawater temperature has been retained as a design basis. The UF section remains essentially unchanged at the higher temperatures.

4.5 Performance Analysis

In order to ensure that the RO system design meets all specified design conditions over the entire range of operating temperatures, without exceeding any design or performance limitations imposed on the RO membranes by the manufacturer, parametric design calculations over a range of operating temperatures are required. Figure 4 shows the potable water production capability at elevated temperatures relative to that at the 18°C ambient seawater temperature. As can be seen from Figure 4, the potable water production capability with 10°C of RO feedwater preheat is about 13% higher than that for ambient seawater temperatures under these conditions.

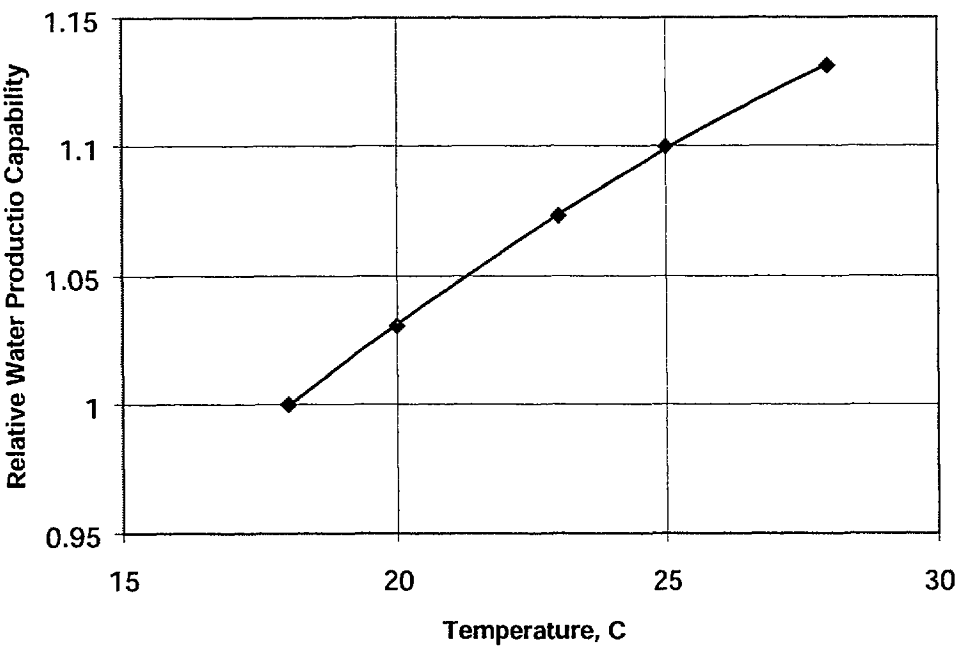


Figure 4
Relative Potable Water Production Capability as a Function of RO System Feedwater Temperature

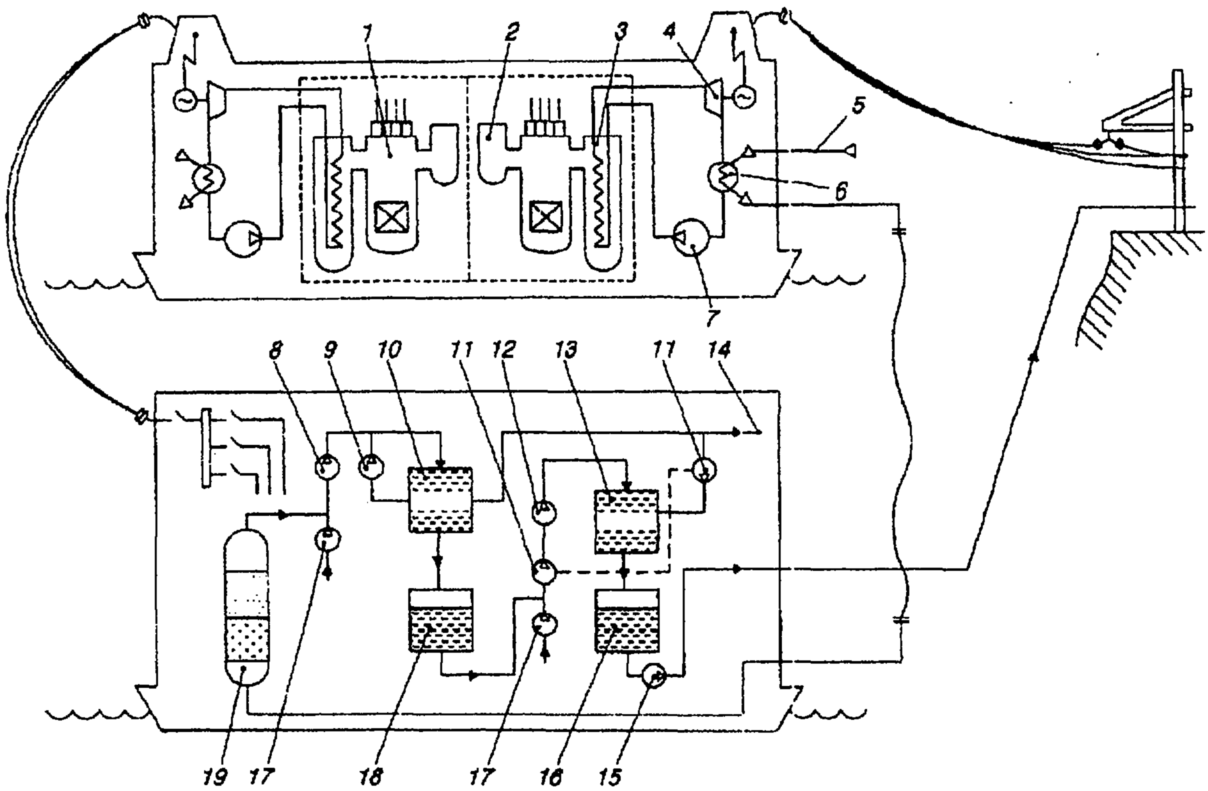
The detailed results from the analysis of RO system characteristics as a function of temperature are presented in Table 1 below.

Table 1
RO System Design Characteristics as a Function of Feedwater Temperature

Temperature, °C	18	20	23	25	28
RO System Feed Flow, m^3/d	233088	233088	233088	233088	233088
Recovery, %	38.4	39.5	41.2	42.2	43.4
Potable Water Flow, m^3/d	89459	92175	96000	98375	101186
Potable Water TDS, ppm	339	357	388	410	442
Energy consumed, MW (approximate)	18.5	18.5	18.5	18.5	18.5
Energy consumed, kW·h/m^3	4.96	4.81	4.63	4.51	4.39
Relative Water Production	1.0	1.030	1.073	1.100	1.131
Relative Energy Consumption	1.0	0.97	0.93	0.91	0.89

Since these calculations are all carried out with the same feedwater flow, the same number of vessels and RO membranes and the same operating pressure, the energy consumption and operating costs are essentially the same for all cases. However the potable water production is increasing significantly with increasing temperature. Hence operation at the higher temperatures results in a reduction in the energy consumption per cubic meter of potable water produced, with a corresponding reduction in the unit cost of potable water production.

A preliminary design concept for the floating nuclear desalination complex has been developed by OKBM and was described by them at an IAEA Advisory Group Meeting on "Floating Nuclear Power Plants for Seawater Desalination" held in Obninsk, Russia, in May 1995[16]. Preliminary peformance analysis for the desalination system were presented[13] at that same meeting. Development has being carried forward as a joint Canada-Russia project, with the FPU based on the KLT-40 shipboard reactor plant and the desalination plant based on the CANDESAL application of reverse osmosis seawater desalination. The two barge concept as it is currently envisaged is illustrated in Figure 5.



1. Reactor
2. Primary circuit circulation pump
3. Steam generator
4. Turbo-generator
5. Seawater intake
6. Condenser
7. Secondary circuit electric pump
8. Medium pressure pump
9. Recirculation pump
10. UltraFiltration membranes
11. Energy recovery system
12. High pressure pump
13. RO membranes
14. Brine discharge
15. Potable water pump
16. Potable water storage tank
17. Anti-scalant injection system
18. Clarified water tank
19. Prefilter

Figure 5
Principle Flow Diagram of the
Floating Nuclear Desalination/Cogeneration Complex

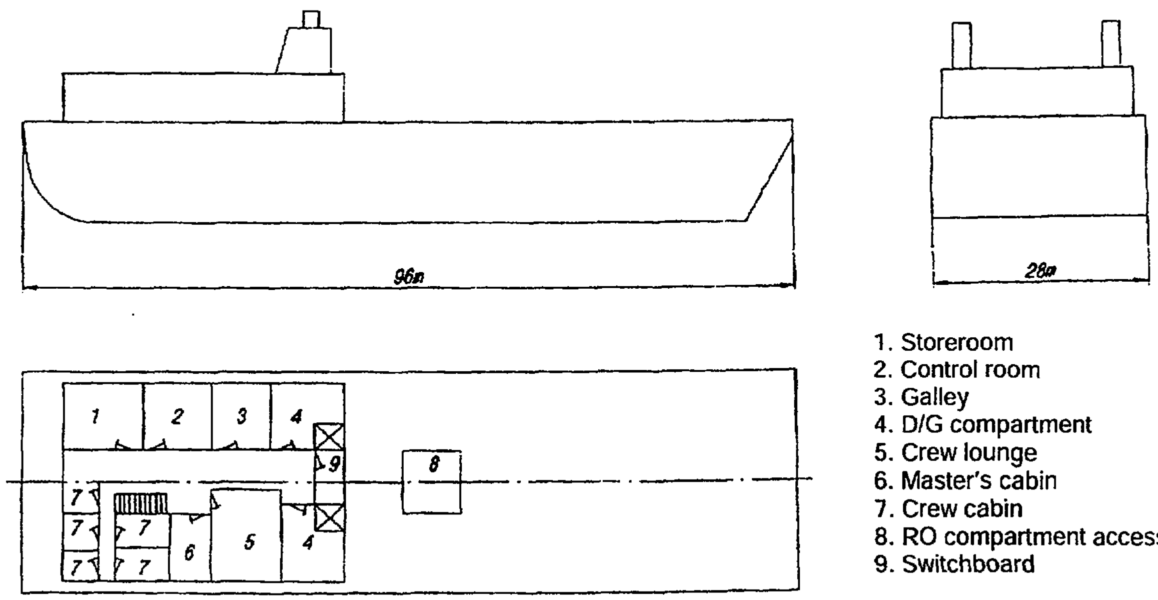
4.6 Conceptual Barge Design

A conceptual design layout for the desalination barge is shown in Figures 6 and 7. The proposed barge design is 96 m long by 28 m wide. The design includes UF capacitance tanks and RO permeate storage tanks below the UF/RO deck. Appropriate baffles and pumping arrangements will be required to restrict free water movement while the barge is under tow. Accommodations, crew spaces and the control room are located on the upper deck.

5. Conclusions

The use of nuclear power as a source of energy for potable water production is both technically viable and economically competitive. CANDESAL's system integration and design optimization techniques provide significant improvements in the efficiency of energy use and the economics of water production. These features will allow nuclear desalination to play an important role in the solution to the growing global demand for water and electricity.

The unique CANDESAL approach can be applied in a floating desalination/cogeneration station based on the KLT-40 reactor and Canadian RO water purification technology. Integrating the reactor and RO systems into a single cogeneration facility allows for a design in which the full benefits of design optimization and system integration, including RO feedwater preheat, can be fully realized. These benefits include reduced plant capital cost, longer RO membrane lifetimes resulting in a reduced membrane replacement requirement, reduced operating and maintenance costs, improved energy efficiency and reduced water production costs. This "marriage" of Canadian and Russian technology leads to improved economics in small-scale nuclear desalination systems. Such systems then become more attractive in developing areas where the requirements for fresh water production are on the order of a few hundred thousand cubic meters per day or less, and where a need for additional electrical generation exists but existing electrical grids can not absorb the supply from larger nuclear generating stations. Finally, the use of floating platforms for the system allows for easy redeployment and hence maximum flexibility for potable water production in regions where population centres are widely scattered and where the water needs vary widely from area to area.



1. Storeroom
2. Control room
3. Galley
4. D/G compartment
5. Crew lounge
6. Master's cabin
7. Crew cabin
8. RO compartment access hatch
9. Switchboard

Figure 6
Reverse Osmosis Desalination Barge

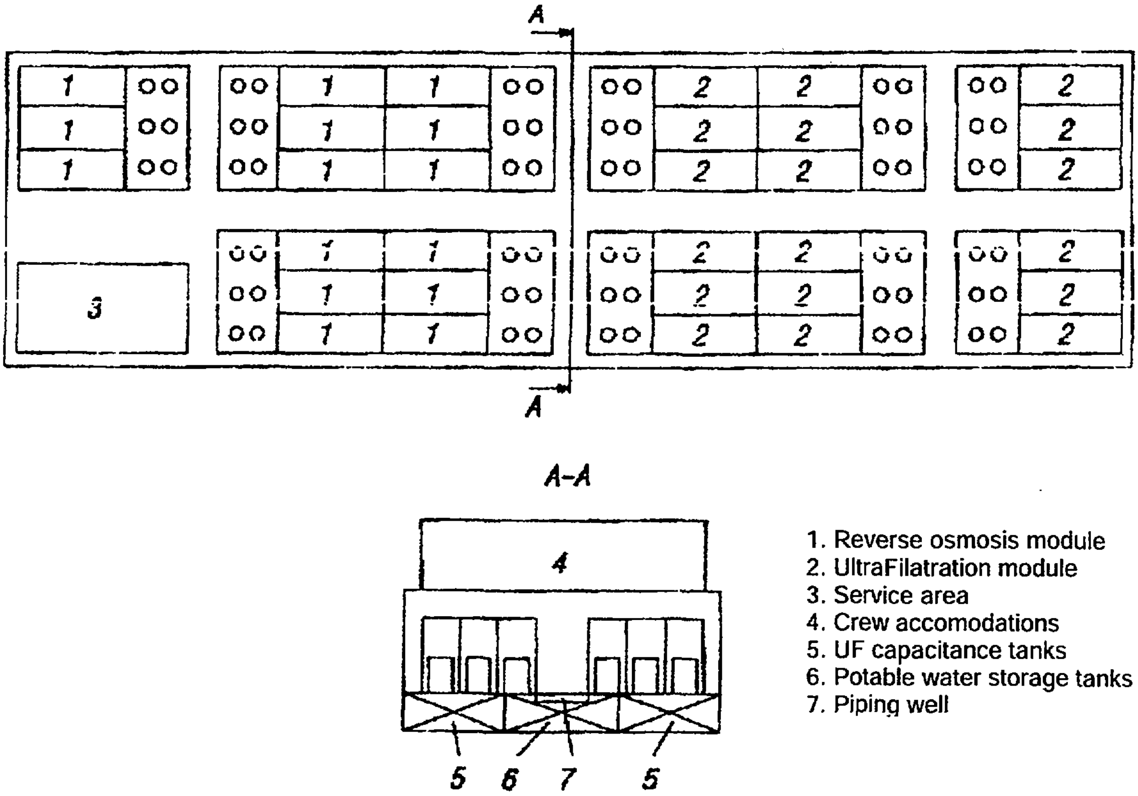


Figure 7
Desalination Barge
Desalination Plant Layout

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IMPLEMENTATION OF THE PROJECT FOR THE CONSTRUCTION AND OPERATION OF A NUCLEAR HEAT AND POWER PLANT ON THE BASIS OF A FLOATING POWER UNIT WITH KLT-40C REACTORS

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Abstract

This paper presents the results of research and development on floating nuclear power plant (FNPP) for electricity and heat production for remote locations and small island or coastal communities. Evaluations of construction period, social and economic factors as well as safety and operational issues of the non-self-propelled barge-mounted NPP is given.

1. INTRODUCTION

The development of small-scale power for isolated regions of Russia has long slipped the attention of the public and specialists. Development of the United Power System of Russia, creation of high capacity generating facilities and construction of giant power plants and complexes have been considered top priorities over the past years.

The decentralized power supply territory occupies about two thirds of Russia. It is mainly populated by small ethnic groups including the peoples of the North whose living standards are dozens of years backward as compared with those of such groups of population in the similar climatic zones of North America and Europe.

On the other hand, this territory is very rich in all kinds of mineral resources which are not mined at the present time as there are no required conditions or infrastructure, we mean power supply, communication and transport, first of all. For this is a very scarcely populated territory the border of which practically coincides with the permafrost zone boundary it is obviously impossible to solve a power development problem by means of large-scale network construction.

As a result of the defense industry conversion a number of small-scale power projects became available for civil applications. These Projects were developed for marine and submarine fleet, spacecraft and other scientific purposes, being notable for their novelty and advanced technological level in comparison with similar projects either in Russia or abroad.

At the present moment according to the IAEA surveys nuclear desalination - i.e. using nuclear reactors as sources of electricity or heat for desalination plants - is considered to be the most promising line of development. The present lack of potable water in some areas and increasingly serious and pressing water supply problem definitely act as a stimulant to the growing display of interest in this field.

According to the international organizations more than 50 countries in the world have to face a critical situation with potable water supply. This refers to the south of Europe, North Africa, East and West coast of the USA, coast of Brazil, Argentina, Chili, some regions of India, China, Egypt, Pakistan, Iran, Mexico. Kuwait, Saudi Arabia, United Arab Emirates, Bahrain, Yemen and other Arabian Peninsula countries, some of Caribbean Sea, Pacific and Atlantic Ocean islands are almost absolutely arid.

The main objective set at the beginning of this decade was to sort out electricity, heat and potable water production projects that are more suitable for decentralized power supply areas.

At the beginning of this process more than 20 small-scale nuclear projects were presented. They were divided into three groups:

- less than 10MW (heat);
- 10 - 50MW (heat);
- over 50MW (heat).

For the first group the winners were identified for heat plants and nuclear heat and power plants (NHPP) separately. "Elena" ranked first among heat plants and got a recommendation for the construction of the first production prototype. "Sakha 2" and "Krot" ranked second among heat and power plants.

For the second group the winners were identified for land-based heat plants, land-based NHPPs and floating NHPPs separately. "Ruta" project ranked first among land-based heat plants. The NHP Ruta has a good potential market in settlements that are connected to the central grid but have no fuel of their own for heat supply. "Angstrem" project ranked first among land-based NHPPs, and ABV 6 ("Volnolom") - among floating ones. It was recommended that the implementation of the latter should be started.

For the third group the winners were identified for land-based and floating NHPPs separately. NHPP 80 ranked first among land-based NHPPs, and KLT-40 - among floating ones. The plant is produced in lots for nuclear icebreakers and have already been examined and accepted by Russian State and international experts.

It was recommended that when selecting the type of a plant for the specific customer the judges of the competition should give preference to floating NHPPs (the winners are presented in Table 1).

TABLE I."SMALL-SCALE NPP-91" PROJECT COMPETITION WINNERS

Capacity	Less than 10MW(heat)		10-50MW(heat)			Over 50MW (heat)	
Plant Type	NHP	NHPP	NHP	Stationary NHPP	Floating NHPP	Stationary NHPP	Floating NHPP
Project	Elena	Cakha-92 Krot	Ruta	Angstream	ABV-6	NHPP-80	KLT-40

In 1995-1996 under the auspices of IAEA an options identification programmer for demonstration of nuclear desalination was completed. As regards small-scale reactors (100-200MW (heat)) for co-generation it was recommended that the power generating desalination complex on the basis of KLT-40C reactors proposed by Russia should be constructed. The above reactors have proved to be safe in operation for more than 150 reactor years.

At the beginning of the 1990s small-scale power plants market research was carried out in Russia as well as abroad.

In the framework of this activity Feasibility Reports for the Chukotsky Autonomous Region, Primorsky and Khabarovsky Regions were prepared. On consideration of small-scale nuclear power for the north of Russia over 250 possible sites were analyzed. The sites were assessed on the basis of the following criteria:

- settlements with very small population were excluded (less than 1000 people);
- settlements having their own fossil fuel were excluded;
- settlements without any potential for future development were excluded;
- settlements with limits related to the natural conditions were excluded;
- priority was given to large developing settlements with a complex and multistage fuel supply scheme.

26 sites were selected for possible location of small-scale nuclear power plants of various types.

For the sites selected a detailed analysis of regional energy markets was carried out, including, among other kinds of studies made for the Chukotsky Autonomous Region, the following:

- present conditions and social and economic development perspectives;
- conditions of natural ecosystems surrounding optional sites;
- electricity and heat consumption;
- technical and financial position of power sector enterprises;
- estimation of fuel resources availability and perspectives of their development;
- the existing plants reconstruction and revamping costs evaluation;
- evaluation of economic efficiency of fossil fuel, renewable resources and nuclear options.

The results of these investigations showed that the implementation of a Project for the construction and operation of a small-scale nuclear heat and power plant on the basis of a floating power unit with KLT-40C reactors (hereinafter referred to as the Project) allows - in the short run and with lower costs - to ensure safe and secure power supply in remote and isolated areas with extreme natural conditions and costly fossil fuel transportation.

Pevek, Chukotsky Autonomous Region, was recommended for the implementation of the pioneer project.

The results of the above work served as a reason for including the Project in the following State Programmes of Russia:

- "Fuel and Energy Programme for 1996-2000".
- "Programme for the Development of the Power Industry of Russia for 1998-2005 and up to 2010".

2. CONCEPTUAL POINTS.

2.1. Plant Description

The small-scale nuclear heat and power plant on the basis of a floating power unit with KLT-40C reactors (SS NHPP on the basis of a FPU with KLT-40C reactors) is designed for generation and supply of heat and electricity to the consumers (Figs 1 and 2).

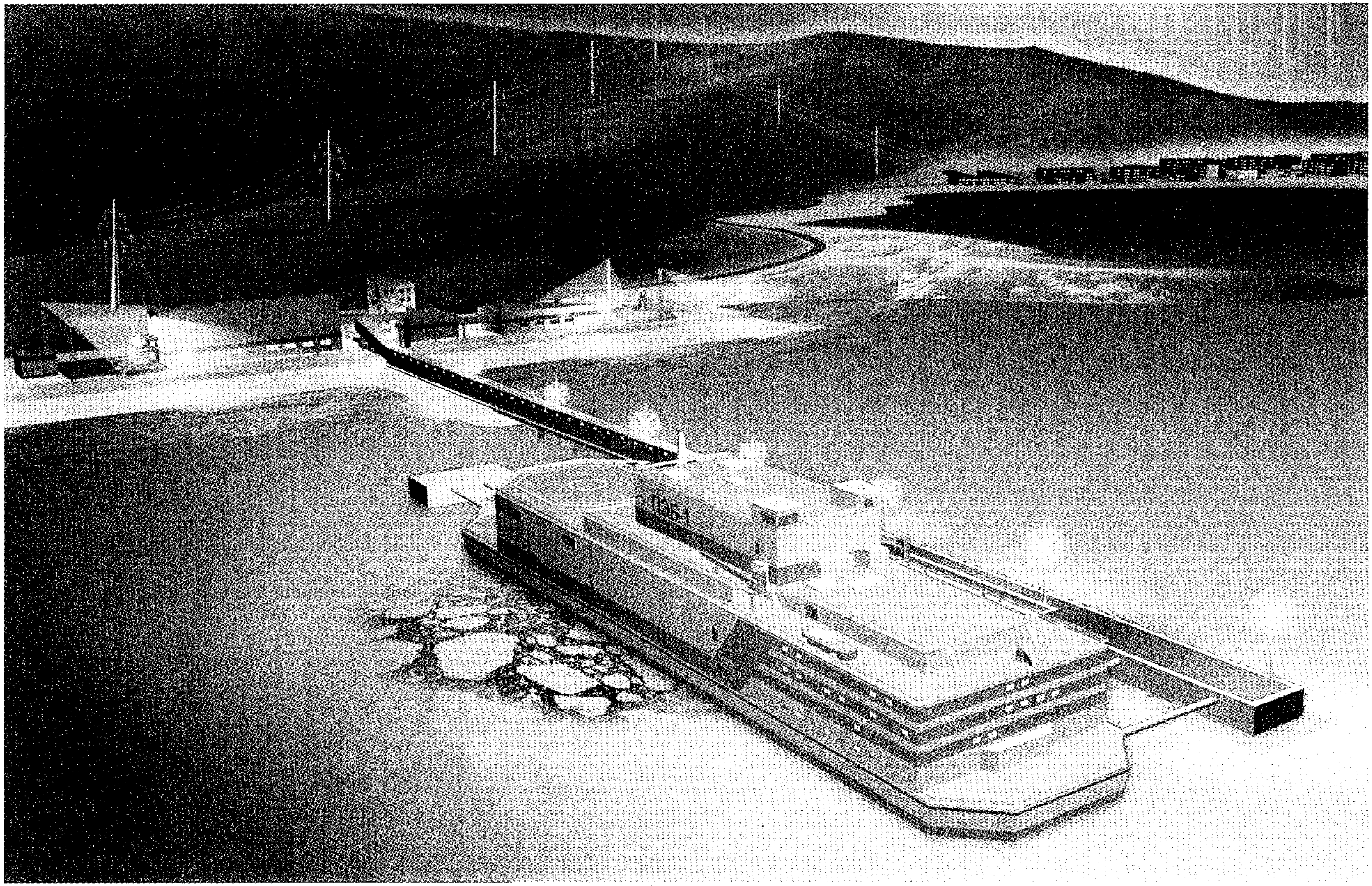

FIG. 1. PEVEK Project from nuclear icebreakers to nuclear power stations.

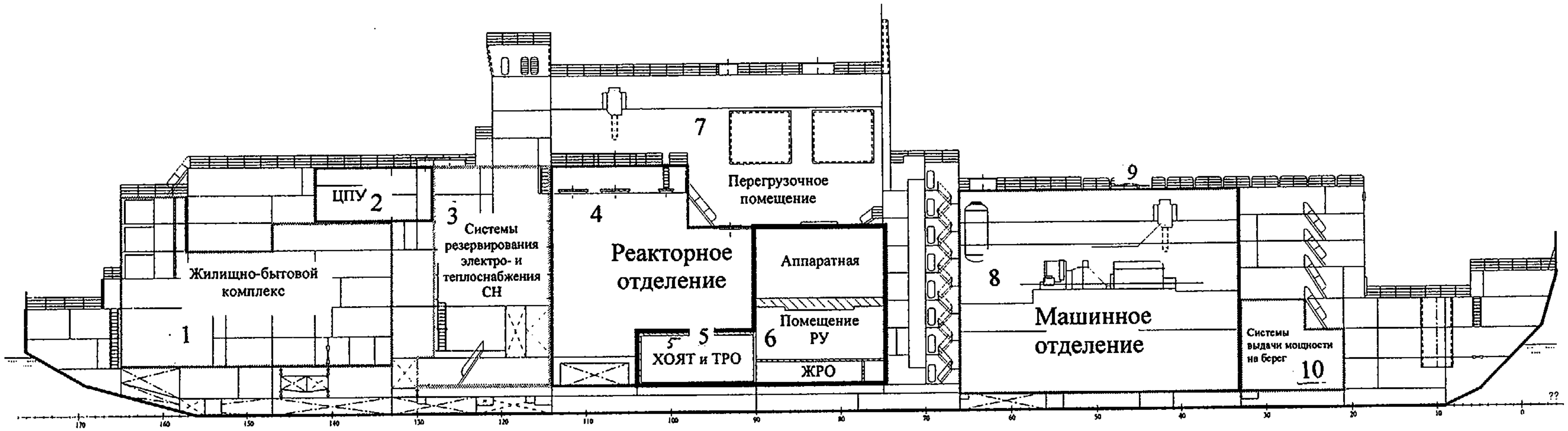

Fig. 2 FPU longitudinal cut-away view:

1 – living section	2 – control room
3 – auxiliary compartment	4 – reactor compartment
5 – spent fuel and waste storage	6 – reactor division
7 – refueling compartment	8 – turbine-generator compartment
9 – helicopter landing site	10 – compartment for electric transformer and switchgear

SS NHPP on the basis of a FPU with KLT-40C reactors consists of a floating power unit, hydraulic structures and a shore-based infrastructure.

The floating power unit is to generate heat and electricity and supply electricity and heat supply water to the shore-based facilities.

The hydraulic structures are designed to connect the FPU to the shore. Mooring facilities provide technical connection between the FPU and the shore. Vessels for supplying the Plant with material and equipment or other maintenance means can be moored to the FPU.

The shore-based structures comprise facilities for the transmission of heat and electricity to consumers and office and other buildings.

Basic characteristics of a small-scale nuclear heat and power plant on the basis of a floating power unit with KLT-40C reactors are given in Tables 2-5.

TABLE II.BASIC CHARACTERISTICS OF A SS NHPP ON THE BASIS OF A FPU WITH KLT-40C REACTORS

Nominal Co-generation Operation	
Nnom, MW	2x35
Qnom, Gcal/hour	2x25
Nominal Heat Generation Operation, Nnom, MW	2x35
Maximum Electric Generating Capacity, MW	2x38,5
Maximum Heat Generating Capacity, Gcal/hour	2x41,8
Auxiliary Electricity Consumption, MW	4-6
Auxiliary Heat Consumption, MW (heat)	3,2
Water into the Heat Supply System, m^3/hour	2x240
Intermediate Circuit Water Temperature Curve for nominal operation, ^0C	130/70
Intermediate Circuit Water Temperature Curve for peak load operation, ^0C	170/70
Water Pressure in the Intermediate Circuit, MPa	1,6
Shore Territory, hectares	1,5
Surrounding Water Area, hectares	6,0

The FPU is being manufactured at a special shipbuilding dock with a very strict control of the quality of production in compliance with the quality assurance procedures established for nuclear vessels. As soon as the required testing is completed the FPU is to be towed to the site complete and ready for operation.

The FPU accommodates two KLT-40C reactor plants with pressurized water reactors, two steam turbine plants with TK - 35/38 - 3.4 turbines of a co-generation type and TAG8123EUL5B electric generators, facilities for nuclear fuel management and solid and liquid radioactive wastes storage, ecological section.

The FPU ecological section is designed for discharge water and oil contaminated water treatment, solid domestic and food waste collection, treatment and disposal.

TABLE III. BASIC TECHNICAL CHARACTERISTICS OF THE FPU

Type	Non-Self-Propelled Vessel
Russia's Register Class	KE✪[2]A2
Length, m	140,0
Width, m	30,0
Hull Height, m	10,0
Draught, m	5,6
Displacement, tons	21 000
Cabins for Personnel, main/reserve	64 (single)/10 (double)
Design Service Life, years	40

TABLE IV.OPERATION OF FPU WITHOUT THE REPLENISHMENT OF:

Nuclear Fuel (time between refueling), years	2,5 – 3
Fossil Fuel (emergencies, transportation), days	30
Fresh Water, days	20
Provisions, days	60

TABLE V.FPU SAFETY PARAMETERS

Total Design Service Life, years	40
Design Lifetime before Overhaul , years	10-12
Docking Periodicity, years	10-12

The Project for KLT-40C reactors provides for new technical solutions as compared to the existing transportable reactor plants:

- system of residual heat removal in case of beyond design basis accidents with complete 24-hour blackout;
- containment excess pressure reduction system with passive bubbling and condensation subsystems;
- emergency core cooling system with a passive subsystem and returning coolant leakage back into the circuit;
- reactor containment cooling system for beyond design basis accidents;
- no-waste technology system;
- direct-acting device for beyond design basis accidents localization.

2.2. Project Concept

MARKETING CONCEPT

- Increase energy sales by introducing a well-balanced tariff policy stimulating the development of mining industry in the Region.
- Effect integration with BiNHPP, FIPP and NPTN in order to establish an optimum dispatching schedule for ChBPS.
- Enter into agreements with large consumers and the authorities of the Region with the object of ensuring timely payments for the energy sold to the consumers (solving the "non-payment" problem).
- Actively develop and expand sales market for the energy generated by the NPP by displacing fossil fuel power generating sources.

COMMERCIAL CONCEPT – Implement the Project using "Build-Own-Operate" scheme, which will allow to remove the heavy burden of intensive capital investment from energy consumers while, at the same time, the Company retains the rights of property.

OPERATING CONCEPT – Operate the NHPP with specially staffed and trained watch crews of highly qualified specialists, which guarantees safety and efficiency of operation and relieves the social burden in the area of NHPP location.

TECHNICAL CONCEPT – At all stages of implementation, such as design and engineering, equipment manufacturing, construction and operation, apply technical solutions proved by many years of trouble-free operation, which will allow to reduce the amount of work, including assembly, installation, erection and repair work at the site, and also to a maximum possible extent assure safe nuclear waste and spent fuel management in order to ensure nuclear and radiation safety of the NHPP.

SOCIAL AND ECONOMIC CONCEPT

– Safe and secure power supply of consumers, reduction of fossil fuel requirements and, as a result, of the volume of fuel transportation under the severe conditions of the Arctic.
– Reduce the need for budgetary appropriations for the power sector, create conditions prerequisite for the development of industrial and agricultural sectors with high potential and simultaneously reduce the number of people employed in coal mining industry, transport and power industry.
– Improve social and ecological conditions in the Region where the Plant is to be located.

2.3. Project Implementation Model

It is proposed that the Project should be implemented using BOO (Build - Own - Operate) model.

Rosenergoatom Concern will finance the construction of the floating power unit, act as the owner and operating organization of the SS NHPP and sell the heat and electricity generated.

A long-term power purchase agreement for selling (buying) energy at a certain tariff will be signed with consumers. The principal provisions of the power purchase agreement - such as the term of validity, tariff rates, risk insurance terms, guarantees etc.- are supposed to be defined when negotiating to agree upon the Feasibility Study results to substantiate the required investment.

The construction period was calculated on the basis of data on the construction of single elements similar to that of the same kind of plants with geographical, climatic, hydrological and other conditions taken into account.

The period of hydraulic structures and shore-based facilities construction stipulated in the Project is 2-3 years.

The main FPU equipment manufacturing period stipulated in the Project was defined on the basis of norms and standards established for nuclear vessels with the following taken into account:

- in prospect - construction of a series of SS NHPPs in order to ensure the continuity of operation of the plant at a certain site;
- time required for manufacturing of certain types of equipment and organization of a continuous production process at the manufacturing plants.

The Plant construction period is 6 years with an allowance for 1 year of assembly, erection, startup and adjustment.

2.4. NHPP Operating Model

The operating model was developed on the basis of design performance characteristics of the complete operating cycle of the NHPP.

The Project provides for the installation of facilities for nuclear refueling and spent nuclear fuel storage on board the FPU without using special service vessels. Radioactive production waste is also stored on board the FPU. Thus, the design autonomous operation period (operation without supplies replenishment) of the FPU is determined by the capacity of spent nuclear fuel and radioactive waste storage tanks and periodicity of docking. With ~0.54 load factor, which corresponds to the pessimistic consumption forecast, the autonomous operation of the FPU is ensured by four nuclear core sets and makes 13-15 years. After the lapse of this period the FPU is replaced with a similar one.

As an FPU can be replaced the NHPP operation can continue as long as you like within the limits of hydraulic structures and shore-based facilities service life.

The following procedures are provided for during the Plant operation:

- 50% two-month load decrease for reactor plants refueling taken in turn every 2-3 years;
- complete interruption of power supply every 13-15 years for FPU replacement.

After the replacement the FPU with full radioactive waste and spent fuel storage tanks is to be towed to specialized repair works at the dock for overhaul, fuel unloading and hull docking.

The overhaul includes the following:

- unloading of spent nuclear fuel from reactor plants and spent nuclear fuel storage facilities;
- complete control of central power compartment equipment and steam turbines and a range of work in order to provide their further operation;
- control of the craft and general ship systems with necessary repair and preventive maintenance;
- FPU hull checkup and repair;
- carrying out a range of work for the FPU to meet the regulatory safety requirements;
- refueling;
- complex testing of all FPU systems and equipment;
- preparing the FPU for transportation to the site.

The overhaul period stipulated in the Project is 1 year with an allowance for transportation time. The Project provides for two overhauls and three operating cycles.

After the completion of the third operating cycle the FPU is to be decommissioned, i.e. towed from the site to the premises of the specialized dock for dismantling and disassembly of ship nuclear equipment.

The practical implementation of the accepted model is ensured by FPU mobility.

The Project technical decisions and the FPU operating model allow to:

- carry out all refueling between overhauls onboard the FPU;
- store all radioactive waste and nuclear fuel onboard the FPU;
- assure high quality of repair work;
- ensure safe and secure FPU decommissioning.

3. CONSTRUCTION PERIOD

The construction period was calculated on the basis of data on the construction of single elements similar to that of the same kind of plants with geographical, climatic, hydrological and other conditions taken into account.

The remoteness of the Region and its severe climatic conditions determine the seasonal character of equipment deliveries and construction work. The period of hydraulic structures and shore-based facilities construction stipulated in the Project, therefore, is 3 years.

The period of the main FPU equipment manufacturing was defined on the analogy of nuclear icebreakers.

The total FPU construction period is 5 years. Transportation of the FPU to the site and startup and adjustment take 0,5÷1 year. The plant construction period from obtaining the construction license is 6 years. With design and development taken into account the investment period makes 7 years.

The approved production and technological NHPP construction scheme consists of three main stages (see Fig.3):

FPU Manufacturing:

- manufacturing of the FPU hull at a specialized shipbuilding yard;
- simultaneous manufacturing of the central power compartment equipment and its sectional assembly as part of the FPU;
- manufacturing and testing of turbine generators at a specialized plant and their subsequent delivery to the manufacturer of the FPU to be assembled and mounted;
- assembly, mounting and testing of the FPU including reaching the design capacity at a shipbuilding yard;
- construction of the hydraulic structures and shore-based facilities on the site before the FPU arrival;
- Putting NHPP into Operation:
- towing the complete and ready FPU to the site;
- assembly, startup and adjustment of the whole complex of the FPU elements together with the shore-based power transmission network;
- putting the Plant into trial-commercial operation.

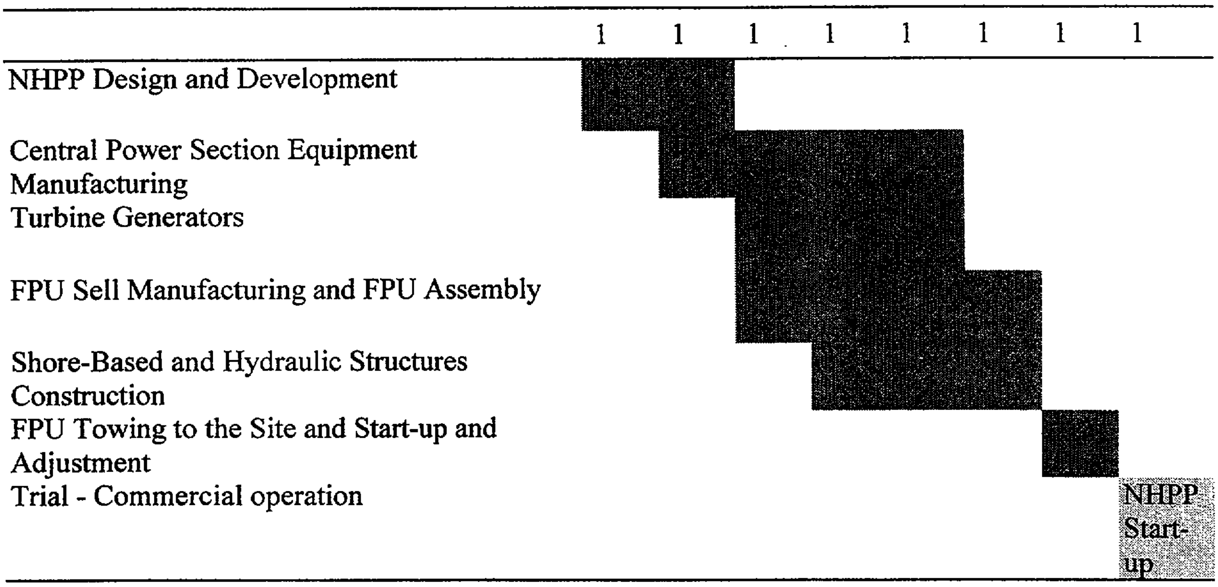

Fig.3. NHPP Production and Technological Construction Scheme

The FPU construction at specialized works in a dock ensures maintaining technological discipline and control of the quality of work carried out at all stages of construction and testing technological cycle.

4. SOCIAL AND ECONOMIC EFFICIENCY OF THE PROJECT

According to the expert estimations, the GDP growth over the construction period will make about 3900 – 5200mln rubles, even under the most pessimistic conditions, i.e. exceed the amount of investment 1,06÷1,42 times.

Creation of additional jobs machine engineering plants and shipbuilding yards in the European part of Russia will bring about GDP growth no less than 4 times over the amount of capital investment.

The construction of a NHPP in Pevek is a key component of the Programme for the Chaun-Bilibinsky Power System. The implementation of the Programme will make a considerable - 2.5-3 times - increase in mining industry output possible, which, in its turn, will bring about the revival of business activity and multiple increase in budget revenues at all levels. The reduction of fossil fuel requirements will relieve the social burden of the ChAR by cutting down the number of people employed in the coal mining and transport industry.

5. RISK ANALYSIS.

The Project sensitivity evaluation shows that the most significant factors affecting the Project parameters are the volume of energy sales, tariffs and construction cost.

- *The energy sales volume* depends on the results of implementation of the Programme for Chukotka gold mining industry development to a considerable extent. The intensification of gold mining and the development of other mineral resources, namely metals, can be considered a top priority not only at the local, but also at the federal level, especially taking into account the Russia's gold and foreign exchange reserve deficit. So the risk associated with the lack of demand for electricity that will be produced by the NHPP is entirely determined by the government policy in this field.

- all the above said equally refers to *tariff rates*, as the local power industry system has no internal market mechanisms to fix prices. Tariffs are established by the authorities. Their level is determined by production and transmission costs and the policy of local authorities. Taking into account the fact that the authorities are interested in the implementation of the Project and ready to participate in financing, tariff policy is anticipated to support and promote the Project in order to improve its technical and economic performance.
- *The cost factor* is of particular importance under the conditions of the economic crisis. On the one hand, price rise makes the Project more costly which can put the main obstacle to its implementation; on the other hand, the same rise is a cause of increase in energy tariffs. Under the conditions of inflation and political instability non-fulfilment of obligations to repay foreign currency credits, but in the given Project this risk is reduced to a minimum as the foreign currency component of construction cost is only 10%.

It should be noted that this Project is to be the first one of a series of floating nuclear power projects. Some of them are to be exported as floating power stations with desalinating plants. Company's activity diversification, getting access to international markets, complex approach to the management of a series of Projects for the construction of a series of floating power plants allow to reduce risks of each single Project considerably.

Thus, the main risk factors are to a certain extent manageable and their possible negative impact upon the Project can promptly be eliminated. The risks will be managed by an organization to be established under the framework of the Programme for the ChBPS Development. This organization is to integrate the interests of large industrial consumers, primary power generating companies, power transmission companies and the authorities of the Region.

6. PROJECT STATUS

At the present time the Feasibility Study for the Project for the construction of a NHPP on the basis of a FPU with KLT-40C reactors in Pevek, Chukotsky Autonomous Region, is completed.

Two materials, namely "A Report on the Present Situation and Conditions of Power Industry in the Chaun-Bilibinsky Power System of the Chulotsky Autonomous Region" and "The Programme for Power Stabilization and Development in the Chaun-Bilibinsky Power System of the Chukotsky Autonomous Region up to 2015", were developed.

The preliminary agreement has been reached on the approval of Pevek site for the NHPP to be constructed.

The Feasibility Study accompanied by the above materials was submitted to the Federal Executive Authorities for expert evaluation, approval and licensing (expert opinion of the Federal Medical and Biological and Extreme Problems has been already obtained).

Declaration of Intention of the construction of a NHPP on the basis of a FPU with KLT-40C reactors in:

- the city of Dudinka, Taymyr (Dolgano-Nenetsky) Autonomous Region;
- the city of Viliuchinsk, Kamchatka Autonomous Region.

During the period May 11 - 21, 1999 an expert of ECTI acting in the framework of TACIS Programme assessed and evaluated the Project. The expert particularly pointed out that the floating application as part of a nuclear power plant benefiting from all the advantages of power generation proved by many years of experience of icebreakers operation can be considered as innovative at the world level, as the first Project of this kind implemented in the world. In case of its successful implementation with all safety and nuclear fuel and radioactive waste management requirements met during the operation the Project will be the first reference to demonstrate the opening new possibilities and perspectives as regards the potential development of isolated and remote northern territories of Russia and foreign countries.

7. CONCLUSION

The Project is oriented towards secure and efficient power supply and offers thoroughly selected optimum solutions of energy supply problem; transparent tariff policy and the consequent solution of the "non-payment" problem; optimization of the existing electricity and heat generation and distribution patterns.

In case of the Project successful implementation final consumers will benefit from the stable and efficient operation of the power supply system and thus, become more confident for paying energy bills. As a result they could be less reluctant to energy saving measures and behavior.

The Project, when implemented with successful operation, will constitute the first world reference for security of energy supply to very isolated areas with extremely severe conditions, allowing stabilization, economic development and social welfare for regions similar to the Chain-Bilibinsky District and ChAO.

Besides, it will also create conditions prerequisite for the efficient use of financial resources, stable operation and efficient performance of the economic systems of the region and local improvement of wealth and welfare.

Replication of this reference Project for new sites will open the market for the key component of the station, which is a small-scale floating nuclear power unit with KLT-40C reactors, reducing production investment and logistics costs that will drop by serial effect for the implementation of the given Project as well as for any further ones of this series.

The Project implemented will allow to reduce fossil fuel consumption fuel transportation costs; reduce the negative impact upon the environment; improve social and economic conditions in the Region; develop the scientific and technical potential of the country; retain the existing and create new jobs in the European part of Russia; expand nuclear industry's importing and international cooperation in the field of nuclear energy use; contribute to improvement of the positive image of nuclear energy for civilian application.

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THE USE OF ENGINEERING FEATURES AND SCHEMATIC SOLUTIONS OF PROPULSION NUCLEAR STEAM SUPPLY SYSTEMS FOR FLOATING NUCLEAR POWER PLANT DESIGN

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Abstract

In the recent years many countries and the international community represented by the IAEA have shown a notable interest in designing small and medium size nuclear power plants intended for electricity and heat generation for remote areas. These power plants can be also used for desalination purposes. As these nuclear plants are planned for the use in the areas without well-developed power grid, the design shall account for their transportation to the site in complete preparedness for operation.

Since the late 80s Research and Development Institute of Power Engineering (RDIPE) has carried out the active efforts in designing the reactor facilities for floating nuclear power plants. This work relies on the long-term experience of RDIPE engineers in designing the propulsion NSSS. Advantages can be gained from the specific engineering solutions that are already applied in the design of propulsion NSSS or from development of new design based on the proven technologies.

Successful implementation of the experience has been made easier owing to rather similar design requirements prescribed to ship-mounted NSSS and floating NPP. The common design targets are, in particular, minimization of mass and dimensions, resistance to such external impacts as rolling, heel and trim, operability in case of running aground or collision with other ships, etc.

DESCRIPTION OF DESIGN FEATURES

2.1.The NSSS is equipped with an integral water-cooled water-moderated reactor with inherent safety and the following unique features:

- negative coefficients of reactivity in the whole operating range of parameters;
- high rate of natural circulation of the coolant which affords effective cooling and heat removal from the core during design-basis and beyond design-basis accidents;
- high heat storage capacity of metal structures and a large mass of coolant in the reactor which result in a relatively slow progression of transients during accidents with upset heat removal from the core.

Figs. 1 and 2 show general view of the reactor and schematic of primary coolant motion. As is clear from the figures, all components of the primary circuit (i.e., the core and control rods, steam generator, main coolant pumps, pressurizer) are located in a single cylindrical vessel.

Primary coolant circulation is provided by main coolant pumps (MCP) with canned asynchronous motors that are installed on the reactor cover. As an additional benefit, simple configuration and short length of the primary circuit path permit to sustain high flow rate of

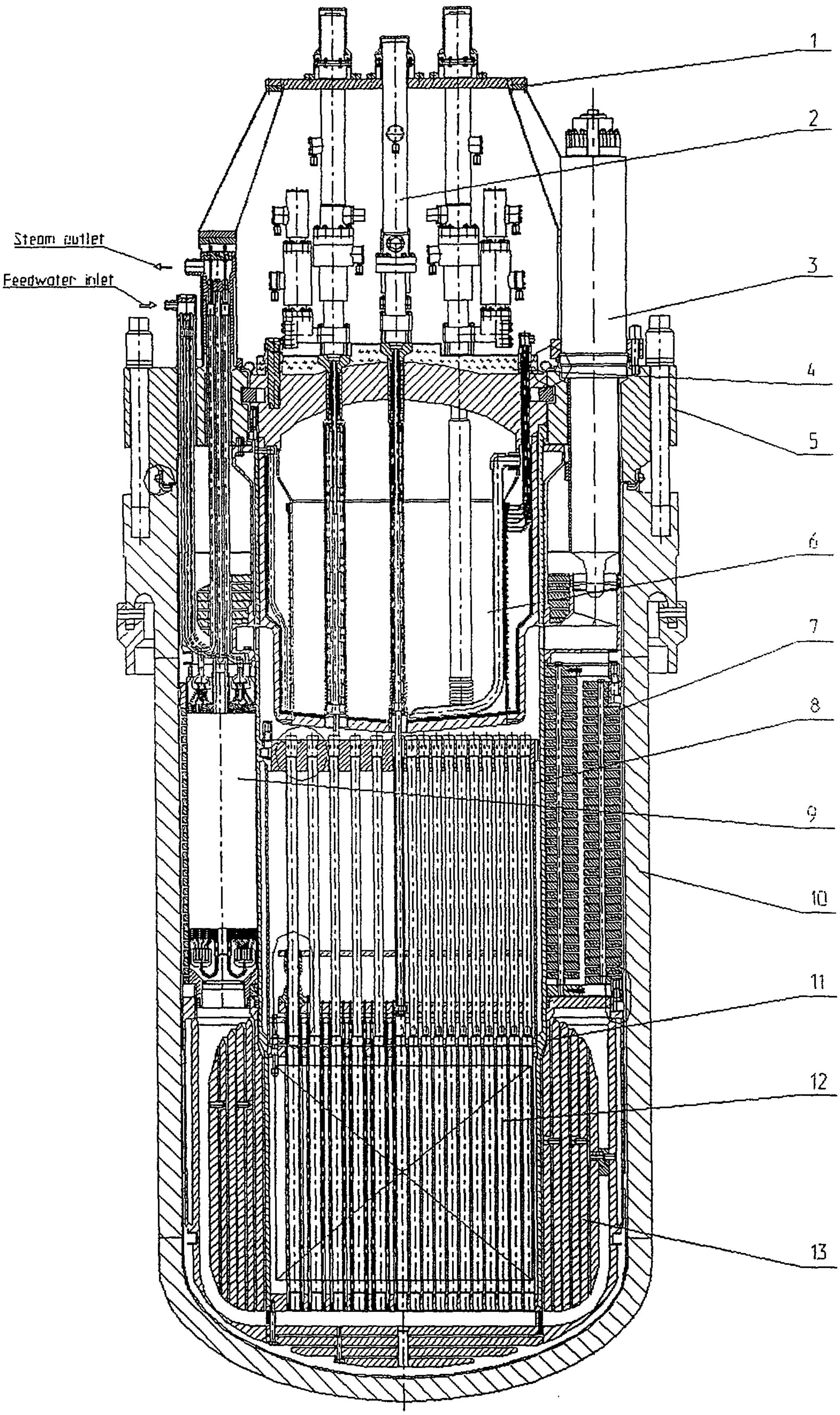

Fig.1. General view of the reactor

1 - drive fastening frame; 2 - shim rod group drive; 3 - MCP; 4 - thermal insulation; 5 - annular cover; 6 - pressurizer; 7 - displacers; 8 - metalwork with control rod clusters; 9 - SG; 10 - vessel; 11 - core barrel; 12 - fuel assembly; 13 - side shield.

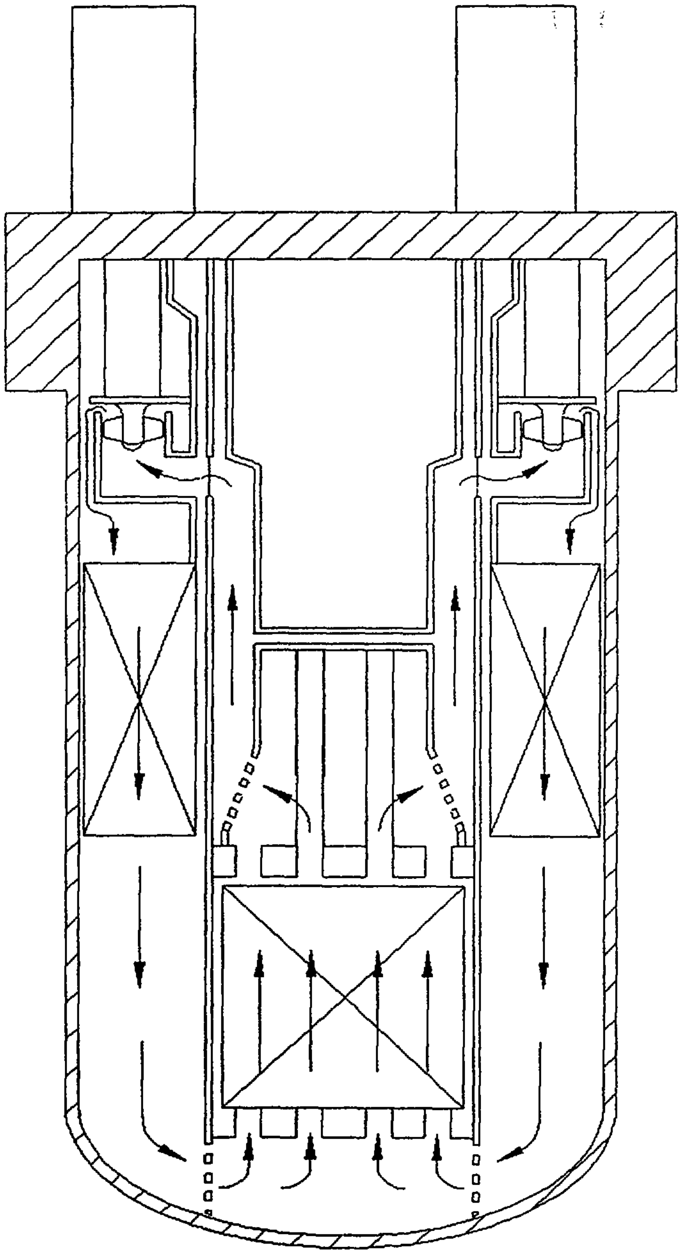

Fig. 2. Primary coolant flow diagram

natural circulation in the reactor and a capability for NSSS operation at power not lower than 25% of nominal when MCPs are stopped.

The reactor core is composed of fuel rods with square cross-section. At the corner there are fins that are spiral with respect to longitudinal axis of the fuel rod. Fuel composition is uranium-zirconium alloy with - 20 % enrichment by U235 . Fuel cladding is made of zirconium alloy. Fuel rods are grouped in fuel assemblies (FA). Burnable rods placed in FAs and absorber rods moving outside fuel channels are used to compensate for reactivity change in the core.

In-vessel once-through steam generator is designed as surface-type helical heat exchanger with tubing made of titanium alloy. Heat exchange area of steam generator is divided into cylindrical cassettes that are placed in the reactor annulus formed by a cylindrical part of the reactor vessel and core barrel. From steam and feedwater sides the SG cassettes are connected via pipelines to form four independent sections that can be isolated by valves outside the reactor vessel.

CEDM incorporates a rotary step motor used for motion of control rods under all normal and emergency modes of NSSS operation. The step motor is backed up with a spring-type actuator that inserts the rods in the core in case of loss of power to the step motor or control system under any position of the reactor, including its capsizing. Implementation of this engineering solution is especially important having in view that the reactor is to be mounted on a ship.

Unlike the known designs of integral reactors being under development in many countries where either steam or steam-gas pressurizer is applied, the integral reactors developed by RDIPE use a gas pressurizer. Selection of such solution was motivated by several reasons, firstly, the intention to simplify and, consequently, enhance safety of the primary circuit pressure compensation system by elimination of heaters and sprinkler system. Secondly, this approach is based on our 40-year experience in designing and operation of ship-mounted NSSS with gas pressurizers in the primary circuit. It should be pointed out, however, that in the previous cases the pressurizers were placed outside the reactor vessel.

In-vessel and out-of-vessel gas pressurizers operate under different operating conditions, especially, temperature conditions. Out-of-vessel pressurizer is in the region of low temperature, therefore water and gas temperatures therein are not higher than 100° C. In-vessel pressurizer that is placed in the upper part of the reactor is exposed to high temperatures. If cooling is not provided temperatures of water and steam-gas mixture will be almost equal to coolant temperature at core outlet. So, instead of gas type this pressurizer will become steam-gas one, which is poorly investigated in operation.

When gas pressurizers are used, it is important to consider gas transport in the primary circuit. It is well known that solubility of nitrogen taken as a working medium in the pressurizer rises as temperature increases. Under 15 MPa nitrogen solubility in water reaches its maximum at temperature of -270° C. As a result, possible gas transport from the pressurizer to the primary circuit, subsequent gas release and formation of gas bubbles in various regions of the circuit will cause certain difficulties as well as some changes in heat transfer conditions in the core and steam generator.

Theoretical justification of in-vessel gas pressurizer design can become the subject of a separate paper. Here we only want to point out that through the use of certain design features in-vessel pressurizer can operate under the conditions and parameters that are similar to those for out-of-vessel pressurizer.

Schematic diagram of in-vessel pressurizer of the integral reactor (Fig. 3) illustrates specific features of its design.

The pressurizer is designed as a cylindrical vessel, its cover being the central cover of the reactor vessel. The pressurizer is separated into two cavities: central where the "water-gas" phase separation level is set in all power modes of operation, and annular peripheral cavity housing a heat exchanger connected to NSSS component cooling system. The annular cavity is connected by pipelines with upper part of the reactor and with the central cavity. Inner surface of cylindrical

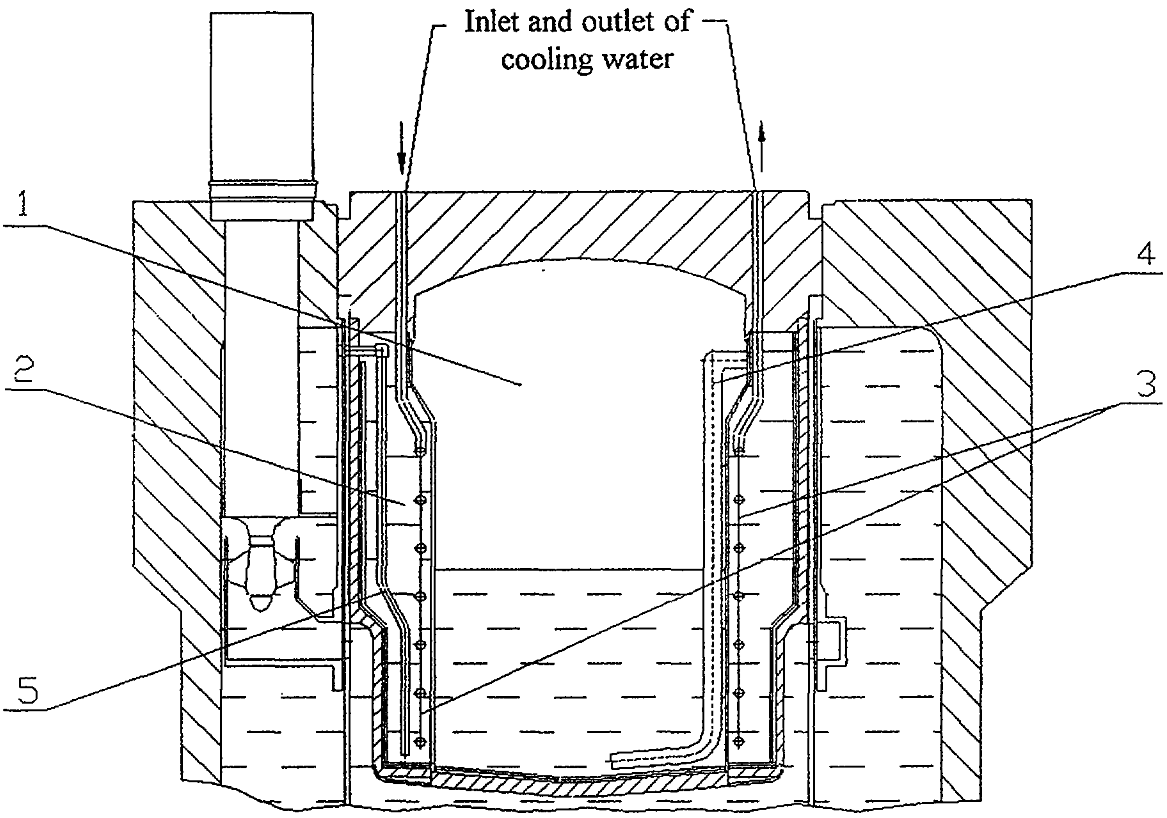

Fig.3. Schematic diagram of the pressurizer

1-central cavity; 2- annular cavity;
3-heat exchanger; 4,5- pipelines

part of the reactor vessel and pressurizer bottom are coated with thermal insulation which is designed as multi-layer set of sheets made of titanium alloy.

When NSSS is in operation, temperature in the central cavity of the pressurizer is settled at ~ 120° C, i.e. these are conditions of minimum nitrogen solubility in water. Besides, sequential connection of the reactor annulus and peripheral and central cavities allows to operate the pressurizer as hydraulic siphon: gas being accumulated in the upper part of the reactor or in peripheral cavity of the pressurizer will flow under fluctuation of the primary temperature to the central cavity and adds to total mass of gas.

Fig. 4 shows arrangement of NSSS components and biological shielding. The figure illustrates a general principle which sets the basis for the design concept of a nuclear power facility. In compliance with this principle, all components of the NSSS are placed in two volumes formed by strong leaktight shells, i.e. the safeguard vessel and containment that act as successive barriers preventing from release of radioactive medium.

Primary coolant components are placed inside the safeguard vessel and under all designbasis accidents the possible release of radionuclides would be retained within the safeguard vessel. Only under beyond design-basis accidents when pressure exceeds the allowable limits the radioactive medium could be released from the safeguard vessel into the containment through the bubbling device.

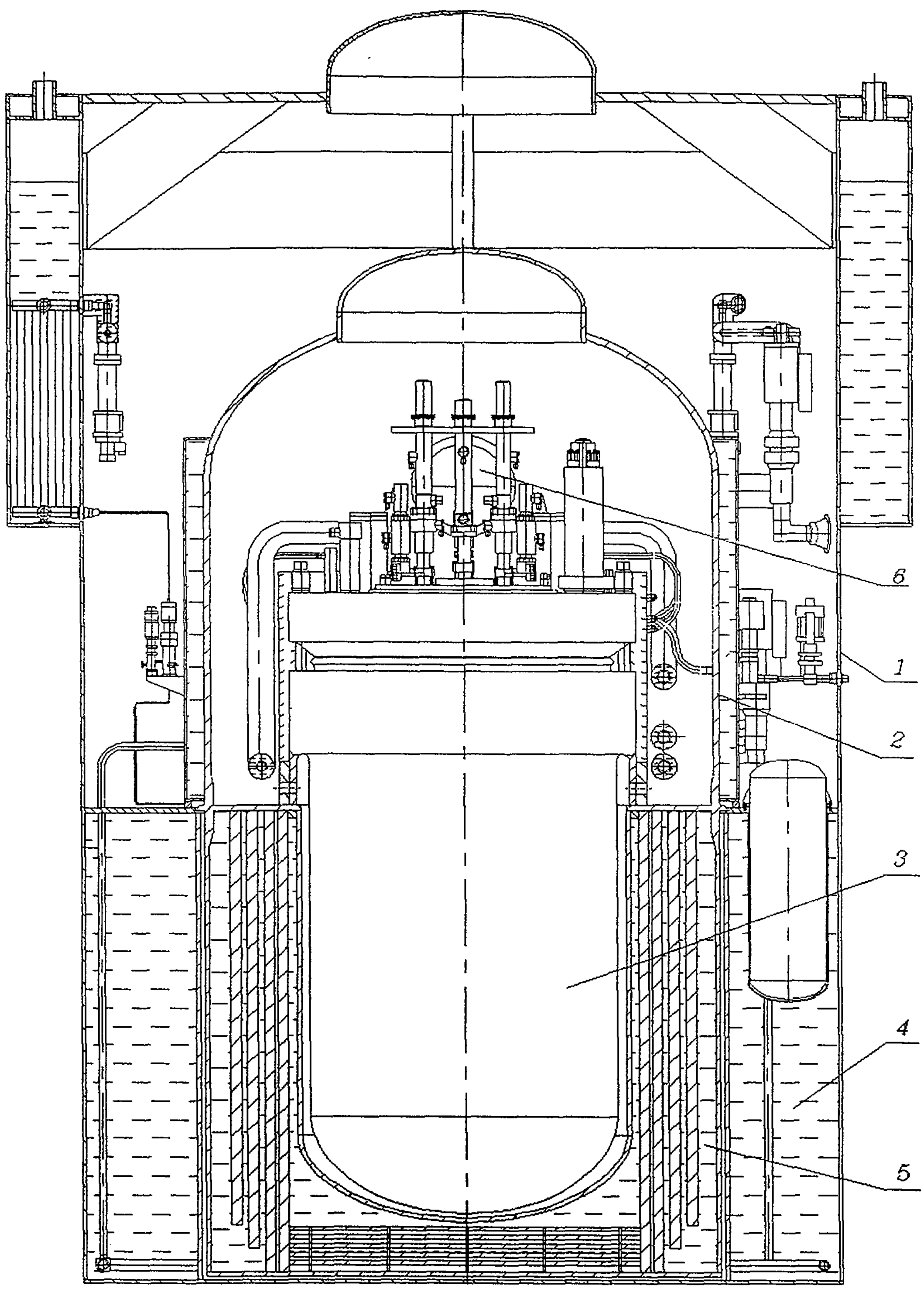

Fig.4. Arrangement of NSSS

1 - containment; 2 - safeguard vessel; 3 - reactor; 4 - biological shielding external tank; 5 - biological shielding internal tank ; 6 - entrance hatch

A regular cylindrical shape of the reactor vessel permits to use the biological shielding optimized in terms of its efficiency and mass and dimension characteristics. Metal-water biological shielding is arranged as two annular concentric tanks. An air gap is provided between the reactor vessel and internal biological shielding tank for the purpose of thermal insulation.

Under the accidents caused by primary coolant leak this gap is filled with water thereby providing adequate heat removal from the reactor vessel. That excludes the probability of reactor vessel meltdown under postulated beyond design-basis accidents involving the core dryout.

Among the safety-related design features it is important to point out the extensive use of passive systems and safety features whose operation is based on natural processes with no need for external power supply.

Such systems include:

- CPS drives whose design assures insertion of control rods into the core by gravity and drop springs;
- passive systems for emergency residual heat removal;
- a safeguard vessel which ensures core coverage with coolant and heat removal under all severe accidents, and guarantees radioactivity confinement in case of a leak in the primary circuit;
- a containment which limits radioactive releases from an open safeguard vessel and under beyond design-basis accidents;
- metal and water biological shielding which apart from its direct functions serves as bubbler tanks with cooling water and provides heat removal from the reactor vessel to avoid its meltdown under a postulated beyond design-basis accident with core dryout.

DESIGN FEATURES OF SMART FOR BARGE MOUNTED APPLICATION

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Abstract

SMART is an integral reactor of 330MWt capacity with passive safety features being developed for a wide range of applications including the barge mounted co-generation plant. Its design strives to combine the firmly-established commercial reactor design with new advanced technologies. Thus the use of the industry proven KOFA (Korea Optimized Fuel Assembly) based nuclear fuels is pursued while such radically new technologies as self-pressurizing pressurizer, helical once-through steam generators, and advanced control concepts are being developed.

The safety of SMART centers around enhancing the inherent safety characteristics of the reactor and salient features include low core power density, integral arrangement to eliminate large break loss of coolant accident, etc. The progression of emergency situations into accidents is prevented with a number of advanced engineered safety features such as Passive Residual Heat Removal System, Passive Emergency Core Cooling System, Safeguard Vessel, Passive Containment Over-pressure Protection. This paper presents the status of current SMART development, characteristics of SMART safety systems and the possibility of SMART application to barge mounted environment.

1. OVERVIEW OF SMART

SMART (System-integrated Modular Advanced ReacTor) is an advanced integral PWR(Pressurized Water Reactor) that produces 330MWt at full power. Major primary components are housed within a single pressure vessel. New, advanced and innovative features are incorporated in the design to provide the reactor with significant enhancements in safety, reliability, performance, and operability. Major design and safety characteristics of SMART can be summarized as follows:

a) inherent safety
b) simplification of systems
c) passive engineered safety systems
d) no large break of loss of coolant accident
e) new generation of man-machine interface systems
f) low core power density
g) large negative moderator temperature coefficient due to soluble boron free core
h) integrated arrangement of primary systems
i) large volume of primary coolant providing large thermal inertia and long response time
j) large volume of passive PZR to accommodate wide range of pressure transients
k) canned motor pumps remove the need of MCP seal
l) low level fast neutron fluence on the RPV
m) passive removal of the core decay heat by the natural
n) The core melt-down frequency is expected to one hundredth of that of conventional reactor.

The design combines firmly established commercial reactor design with new advanced technology. Thus substantial part of the technology and design features of SMART has already been proven in the industries, and new innovative features will be proven through various tests.

Despite disadvantages in the power production cost due to the small power output, SMART can derive the economical advantages from ;

a) simplified system with reduced number of pumps and valves, piping, instrumentation and wiring, etc.
b) flexible operation
c) modularized components
d) on-shop fabrication of components
e) better match to the grid
f) low financial risks, etc.

The overall design parameters of the SMART is shown in Table 1. SMART is expected to fully satisfy the Korean as well as the international safety and licensing requirements. The program for the SMART and its application system development is being carried out jointly by KAERI (Korea Atomic Energy Research Institute), and several nuclear organizations and universities in Korea.

2. SMART REACTOR ASSEMBLY

SMART is an integral type reactor and all of the major primary system components such as fuel and core, twelve steam generators (SG), pressurizer (PZR), four main coolant pumps (MCP), and forty-one control element drive mechanisms (CEDM) are housed in a single pressurized reactor vessel (PRV). The section view diagram of SMART reactor assembly is shown in Fig. 1. With this integral arrang-ement, there is no large size pipe connection and thus no possibility of large break loss of coolant accidents. The primary coolant flows up through the core and then through the MCPs to enters the shell side of SG from the top. The secondary side feedwater enters tube side of the helically coiled SG tubes from the bottom and flows upward. The heat is removed from the primary and superheated steam exits the SGs. Large volumes at the top part of RPV constitute the pressurizer. The pressurizer volumes are occupied by water, steam and nitrogen gas. The pressure is determined by the the partial pressures of steam and gas. These pressures vary in correspondence to the change in the core exit temperature and thus to reactor power. With appropriate combination of MTC and pressurizer size and conditions, the pressurizer can self-regulate the pressure at a desired level without any active control. Twelve SG cassettes are situated at equal-spacing in the annlus region between the RPV and support barrel. To provide sufficient driving force for the natural circulation of the coolant, SGs are located relatively high above the core. This design feature and the low flow resistance endows the system with 25% full power operation capability with natural circulation. The internal shieldings surrounding the core at sides and bottom reduces the neutron fluence of the RPV.

2.1. Fuel and Reactor Core

SMART core consists of fifty-seven (57) fuel assemblies which are based on the industry proven U7xU7 square array Korea Optimized Fuel Assembly (KOFA). Each fuel assembly holds 264 fuel rods, 2U guide tubes for control rods, and 4 instrumentation thimbles. A fixed incore instrumentation is located in one of the 4 thimbles. 5 space grids hold the fuel rods in position. Top and bottom spacer grids are made of inconel, and the 3 middle space grids are made of zircaloy. Specially designed bottom end piece offers improved resistance to the debris entering the core. 4.95 w/o enriched uranium oxide fuel is enclosed in zircaloy clad and has sufficient reactivity for three year or longer operation cycle. The fuel assembly is designed to accommodate power ramps during load-following maneuvers.

Table 1 SMART Design Data Information

GENERAL INFORMATION	
Reactor Name	SMART
Reactor Type	Integral PWR
Thermal Power (MWt)	330
Electric Power (MWe)	100
Design Life Time (yr)	60
FUEL AND REACTOR CORE	
Fuel Type	17x17 Square FA
Active Fuel Length (m)	2.0
Fuel Material	Low enriched UO_2
No. of Fuel Assembly	57
Core Power Density (w/cc)	62.6
Refueling Cycle (yr)	> 3
REACTIVITY CONTROL	
No. of Control Element Banks	41
No. of Control/Shutdown Banks	9/32
No. of Absorber Elements per CEDM	21
Material of Control Banks	Ag-In-Cd
Material of Shutdown Banks	B_4C
Burnable Poison Material	Al_2O_3-B_4C & Gd_2O_3-UO_2
REACTOR PRESSURE VESSEL	
Overall Length (m)	9.8
Outer Diameter (m)	3.96
Average Vessel Thickness (mm)	19.8
Vessel Material	SA508, CL-3
REACTOR COOLANT SYSTEM	
Cooling Mode	Forced Circulation
Total Coolant Mass (kg)	46320
Design Pressure (MPa)	17
Operating Pressure (MPa)	15
Core Inlet Temperature (°C)	270
Core Outlet Temperature (°C)	310
STEAM GENERATOR	
Type	Helically-coiled once-through
No. of SG Cassettes	12
Tube Outer Diameter (mm)	12
Feedwater Pressure (MPa)	5.2
Feedwater Temperature (°C)	180
Steam Pressure (MPa)	3.0
Steam Temperature (°C)	274
Superheat (°C)	40
PRESSURIZER	
Type	Self-controlled
Total Volume (m^3)	21.7
CONTROL ELEMENT DRIVE MECHANISM	
Type	Step Motor Driven
No. of CEDM	41
Design Pressure (MPa)	17
Design Temperature (°C)	350
Moving Distance per Pulse (mm)	0.208
Moving Speed (mm/sec)	0 - 15
MAIN COOLANT PUMP	
Type	Glandless Canned Motor Pump
No. of MCP	4
Design Pressure (MPa)	15
SECONDARY SYSTEM	
Main Steam Flow Rate (kg/hr)	555,120
Feedwater Pressure (MPa)	5.0
Type of Feedwater Pump	Multi-stage
No. of Feedwater Pump	3
Type of Startup Pump	Multi-stage
No. of Startup Pump	2
Type of Condenser	Shell and Tube
No. of Turbine	1 Main and 2 Aux.
No. of Condensate Pump	2
MAKE-UP SYSTEM	
No. of Trains	2
Volume of Make-up Tank (m^3)	2
Design Pressure (MPa)	17
EQUIPMENT COOLING SYSTEM	
No. of Train	1
Coolant Pressure (MPa)	0.5
Coolant Temperature (°C)	40
CONTAINMENT	
Type	Pressurized concrete with steel lining
Design Pressure (MPa)	0.3
Design Temperature (°C)	120

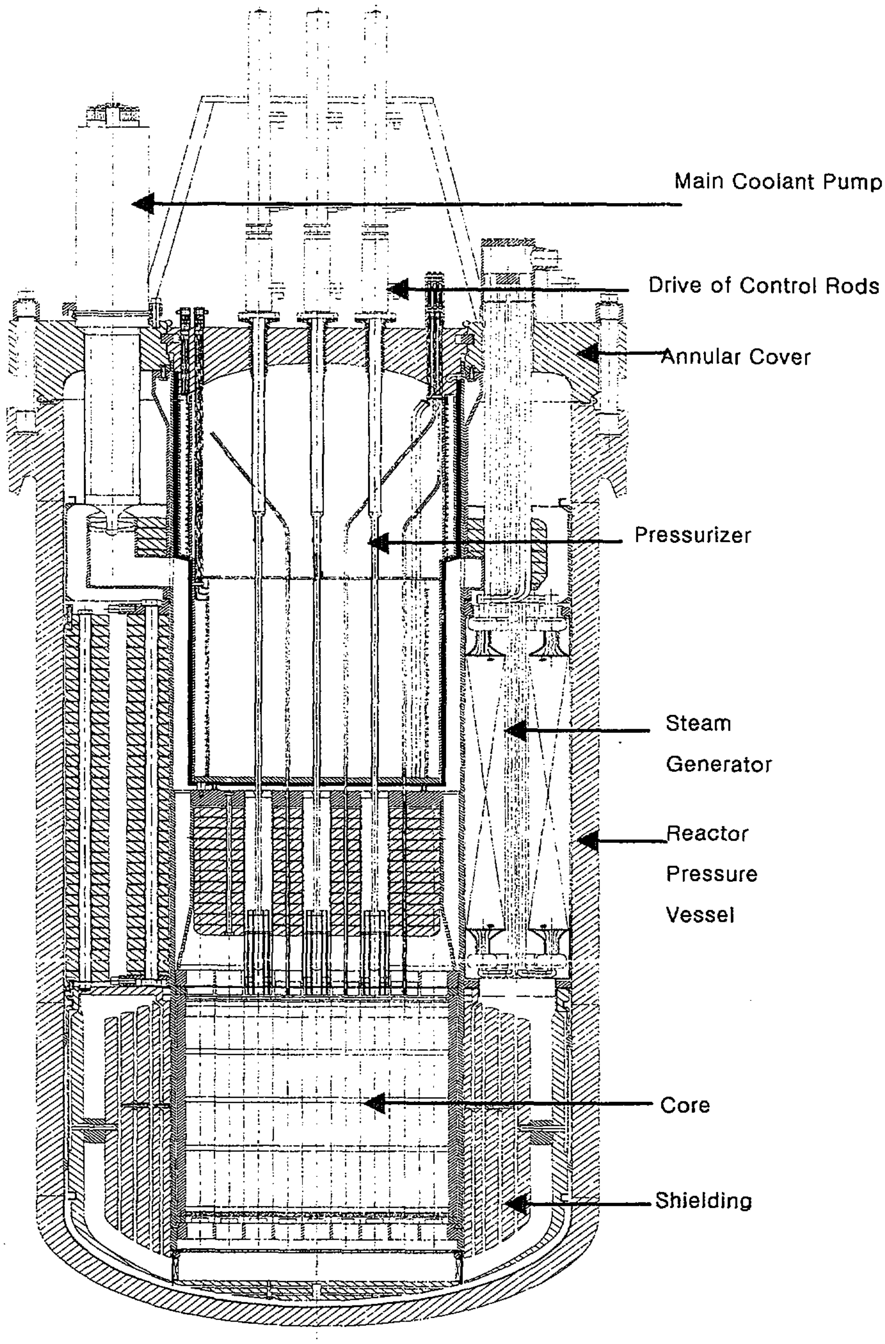

Fig. 1 SMART Reactor Assembly

The SMART core design is characterized by ultra long cycle operation with single or modified single batch reload scheme, low core power density, soluble boron-free operation, enhanced safety with strong negative MTC feedback at any time during the cycle, adequate thermal margin, inherently free from xenon oscillation instability, and minimum rod motion for the load follow with coolant temperature control. SMART fuel management is designed to achieve maximum cycle length between refueling. A simple single batch refueling scheme returns a cycle of 985 Effective Full Power Days (EFPD). This reload scheme minimizes complicated reload design efforts. A modified single batch scheme with 20 peripheral assemblies reloaded at every even numbered cycle is also possible, and thus enhance the fuel utilization. SMART fuel management scheme is highly flexible to meet the customer requirements.

2.2. Reactor Pressure Vessel

The SMART reactor pressure vessel (RPV) is a pressurized cylindrical vessel accommodating all major components of the primary system. The RPV consists of a cylindrical shell with an elliptical bottom and an upper flange part welded up to the shell. The RPV is closed at the top by the annular peripheral cover and the round central cover which also serve as the cover of the in-vessel pressurizer. The annular cover is fixed onto the vessel flange by means of stud bolt joint. The vessel-to-annular cover joint is made leaktight by a welded torus sealing. The central cover is fastened to the annular cover by a flangeless joint, using rack-and-gear mechanism. The annular-to-central cover joint is also made leaktight by a welded torus sealing. All penetrations to RPV are limited in the vessel head region. This assures that the RPV forms a leaktight container under any postulated pipe break accident. On the annular cover, there are twelve SGs, steam collecting and feedwater-distributing chambers, four MCPs, makeup piping nozzles, resistance thermometers, branch pipes, etc. On the outer surface of the central cover, there are nozzles for 4U CEDMs, rack- and-gear drives, branch pipes, etc. The core barrel with fuel assemblies and shielding tube assemblies are located in the lower part of the RPV.

2.3. Steam Generators

SMART has twelve identical SG cassettes which are located in the annulus formed by the RPV and the support barrel. SG cassette is of once-through design with helically coiled tubes wound around the central mandrel. The primary reactor coolant flows downward in the shell side of SG tubes, while the secondary feedwater flows upward in the tube side. Steam at 3.0MPa and superheated by 40°C exits the SG. For the performance and safety aspects, each SG cassette consists of six identical and independent modules with the same number of tubes per module. Each module has one feedwater intake header and one steam outlet header. Six modules from three adjacent SG cassettes - two modules per cassette are then joined into one nozzle, and three nozzles are joined to form one section. This hierarchyal grouping of SG is designed to minimize the impact of SG tube rupture accident on the reactor system. To prevent hydrodynamic instability in the parallel connected tubes, throttling devices are installed at the feedwater inlet of each helically coiled tube. Each throttling device is located in the tube sheet of the feedwater headers, and it is a sleeve with a core connected by a thread joint. Throttling orifices are also installed in the lower part of each cassette on the primary side to provide uniform primary coolant flow rate to each cassette. In the case of normal shutdown of the reactor, the SG is used as the heat exchanger for the passive decay heat removal system (PRHRS).

2.4. Pressurizer

The PZR is located inside the upper part of the RPV, and it is filled with water, steam, and nitrogen gas. The PZR is connected to the gas tank located outside the RPV, where high pressure nitrogen gas is supplied. The primary system pressure is determined by the sum of the partial pressures of nitrogen gas and steam. The pressure in the primary system of is automatically and passively regulated by the thermo-dynamic interactions of water/steam and gas in the pressurizer. The PZR design eliminates the complicated control and maintenance requirements and there is no active pressure control mechanism such as spray or heater. To prevent a relatively large variation in pressure caused by the power change, SMART employs a method for keeping the average primary coolant temperature and PZR temperature constant. The large pressure variation which may occur during power maneuvering is reduced by maintaining the constant temperature of the gas-steam space low and insensitive to the core outlet temperature variation. For this purpose, a PZR cooler is installed to maintain low PZR temperature and a wet thermal insulator is placed between the PZR and the primary system to reduce conductive heat transfer.

2.5. Control Element Drive Mechanism

The SMART Control Element Drive Mechanism (CEDM) is designed for fine-step movement and consists of a linear pulse motor (LPM), choke rack with top and bottom limit switches which also act as control element assembly (CEA) position indicator, hydro-dampers, locking device, and extension shaft connecting CEDM and CEA. The CEDM movement is accomplished with a combination of four phases. As a control command comes to the LPM control unit, the LPM phase I gets off and phase II gets on. After that, phase II gets off and phase III gets on, etc. The direction of the armature movement depends on the communication sequence of the LPM phases. The normal travel length of the control rod is 4mm per pulse, and the travel speed is in the range of 0 - 50 mm/sec. The drives separate unit is joined to each other on the flanges with studs and nuts. The drives are located on the reactor central cover and fastened to the cover by means of flange joints with studs and nuts. Sealing is provided with the use of copper gaskets rectangular in section.

2.6. Main Coolant Pump

The SMART MCP is a canned motor pump which does not require the need of pump seals. This characteristics basically eliminates a small break Loss of Coolant Accident associated with the pump seal failure which becomes one of design bases events in the reactors using conventional pump. SMART has four MCPs vertically installed on the top annular cover of the RPV. Each MCP is an integral unit consisted of canned asynchronous three-phase motor and an axial-flow single- stage pump. The motor and the pump are connected by a common shaft rotating on three radials and one axial thrust bearing. The bearings use a specialized graphite-based material, and the axial bearing performs the function of sealing. The cooling of pumps is accomplished with the component cooling water. The rotational speed of the pump rotor is controlled by a sensor installed in the upper part of the motor. To avoid the reverse rotation of the pump rotor, an anti-reverse device is installed at the motor shaft near the middle radial bearing.

3. SAFETY SYSTEMS

Besides it is inherent safety characteristics of SMART, further enhanced safety is accomplished with highly reliable engineered safety systems. The engineered safety systems designed to function passively on the demand consist of reactor shutdown system, passive residual heat removal system, emergency core cooling system, safeguard vessel, and containment overpressure protection system. Additional engineered safety systems include the reactor overpressure protection system and the severe accident mitigation system. The schematic diagram of the SMART safety system is shown in Fig. 2.

3.1. Reactor Shutdown System

The shutdown of SMART can be achieved by a function of one of two independent systems. The primary shutdown system is 32 shutdown banks of CEA of which absorbing material is B_4C. The control banks are dropped into the reactor core by the gravity force and immediately stops the neutron chain reactions. These control banks have sufficient shutdown margin to bring the reactor from hot full power to hot shutdown, even with a most reactive bank stuck out of the core. For the case of failure of the primary shutdown system, the emergency boron injection system is provided as a backup system and consists of two tanks, $6m^3$

each filled with 30 g of boric acid per Ukg water. One train is able to bring the reactor to the subcritical state. The system is an active system working with pump.

3.2. Passive Residual Heat Removal System (PRHRS)

The system passively removes the core decay heat and sensible heat by natural circulation in case of emergency such as steam extraction, unavailability of feedwater supply,

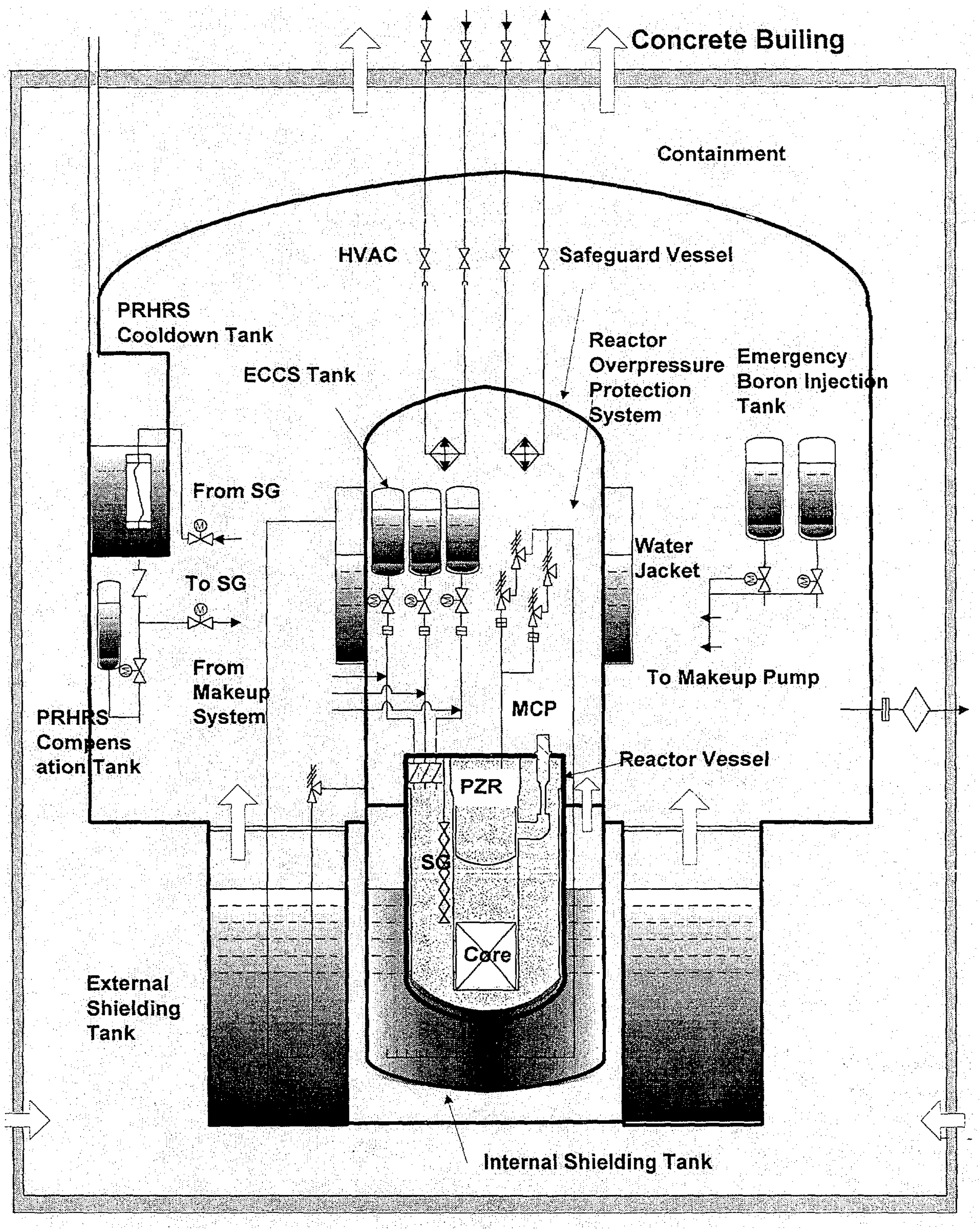

Fig. 2 SMART Safety Systems

and station black out. Besides, the PRHRS may also be used in case of long-term cooling for repair or refueling. The PRHRS consists of 4 independent trains with 50% capacity each. Two trains are sufficient to remove the decay heat. Each train is composed of an emergency cooldown tank, a heat exchanger and a compensating tank. The system is designed to keep the core un-damaged for 72hrs without any corrective actions by operators at the postulated design basis accidents. In case of normal shutdown of SMART, the residual heat is removed through the SG to the condenser with turbine bypass system.

3.3. Emergency Core Cooling System (ECCS)

The SMART design excludes any possibility of large break Loss of Coolant Accident. The largest size of pipes connected to the outside of the RPV is 20mm. The ECCS is thus provided to protect the core uncovery by mitigating the consequences of design basis events such as small break LOCA through make-up of the primary coolant inventory. When an initiating event occurs, the primary system is depressurized. The pressure difference between the primary system and the ECCS breaks the rupture disc installed in the pipe of ECCS and the water immediately comes into the core by the gas pressure. The ECCS consists of three independent trains with U00% capacity each. Each train includes a cylindrical water tank of $5m^3$ in volume pressurized with nitrogen gas, isolation and check valves, rupture disc, and a pipe of 20mm in diameter connected to RPV.

3.4. Safeguard Vessel

The safeguard vessel is a leaktight pressure retaining steel-made vessel intended for the accommodation of all primary reactor systems including the reactor assembly, pressurizer gas cylinders, and associated valves and pipings. The primary function of the safeguard vessel is to confine the radioactive products within the vessel and thus to protect any primary coolant leakage to the containment. The vessel also has a function to keep the reactor core undamaged during 72hrs without any corrective actions at the postulated design basis accidents including LOCA, with the operation of the PRHRS and ECCS. The steam released from the opening of the relief valve of the safeguard vessel at the postulated beyond design basis accidents is sparged into the external shielding tanks and immediately condensed.

3.5. Containment Overpressure Protection System (COPS)

The containment is a steel structure with concrete building enclosing the safeguard vessel to confine the release of radioactive products to the outside environment under the postulated beyond design basis accidents relating to the loss of integrity of the safeguard vessel. At any accident causing the temperature rise and thus the pressure rise in the containment, the containment cooling is accomplished, in the passive manner, by removing the heat from the containment. The heat is removed through the steel structure itself and through the emergency cooldown tanks installed inside the containment. A rupture disc and a filtering system are also provided in the containment to protect the steel structure from overpressure and to purify the released radioactive products at the postulated beyond design basis accidents.

3.6. Reactor Overpressure Protection System (ROPS)

The function of the ROPS is to reduce the reactor pressure at the postulated beyond design basis accident related with a control system failure. The system consists of two parallel trains which are connected to the PZR through a single pipeline. Two trains are also combined to a single pipeline connected to the internal shielding tank. Each train is equipped with a rupture disc and two relief valves. At the postulated beyond design basis accidents, the rupture disc is broken and the relief valve is open by the passive means of flow thrust.

The steam is then discharged into the internal shielding tank through the sparging device and condensed.

3.7. Severe Accident Mitigation System (SAMS)

The function of the SAMS is to prevent the egress of molten corium resulting from a severe accident out of the containment. The egress of corium can be avoided due to the design characteristics of safeguard vessel and containment together with the operation of the safety systems. A small air gap under the reactor pressure vessel (RPV) is filled with water from the Makeup system at the severe accident. The in-vessel cooling prevents the egress of the corium out of the RPV. In addition, the water in the internal shielding tank provides the external cooling of the RPV and prevents the egress of the corium out of the RPV. Hydrogen igniters are provided in the safeguard vessel to remove the explosive hydrogen generated during the severe accident.

4. SECONDARY SYSTEM

The secondary system of the plant with SMART has the same function of removing the heat produced from the primary system, as of the conventional nuclear power plant. However, the secondary system generates superheated steam from the feed water. The system consists of in-vessel helically coiled steam generator, main steam and feedwater system, turbine generators and associated pipings and valves. The system is divided into four independent sections arranged in such a way to minimize the heat removal unbalance in the vessel when one section fails. The turbine generators consist of a main turbine generator and two auxiliary turbine generators. One auxiliary turbine generator is standby and the other is in operation to supply the house loads, while the main turbine is used for the offsite power supply. The pressure of main steam line is always constant during power change transients. The load change is done by the change of feedwater flow rate to the steam generator with the change rate of 5%/min, or by the turbine bypass system when the load change is done immediately. The schematic diagram of the SMART safety system is shown in Fig. 3.

5. AUXILIARY SYSTEMS

The major auxiliary systems of SMART consist of equipment cooling system (ECS) and make-up system. The function of ECS is to remove the heat generated in the MCPs, CEDMs, PZR, and the internal shielding tank. The feedwater supplied from the condensate pump of the turbo-generator is used as coolant to remove the heat. The make-up system of SMART performs the following functions; fill and make-up the primary coolant in case of the primary system leak, supply water to the compensating tanks for the PRHRS and supply water to the gas filling tank during the scheduled shutdown processes. The make-up system consists of two independent trains, each train with one positive displacement makeup pump, a makeup tank, and piping & valves. The schematic diagram of the SMART safety system is shown in Fig. 4.

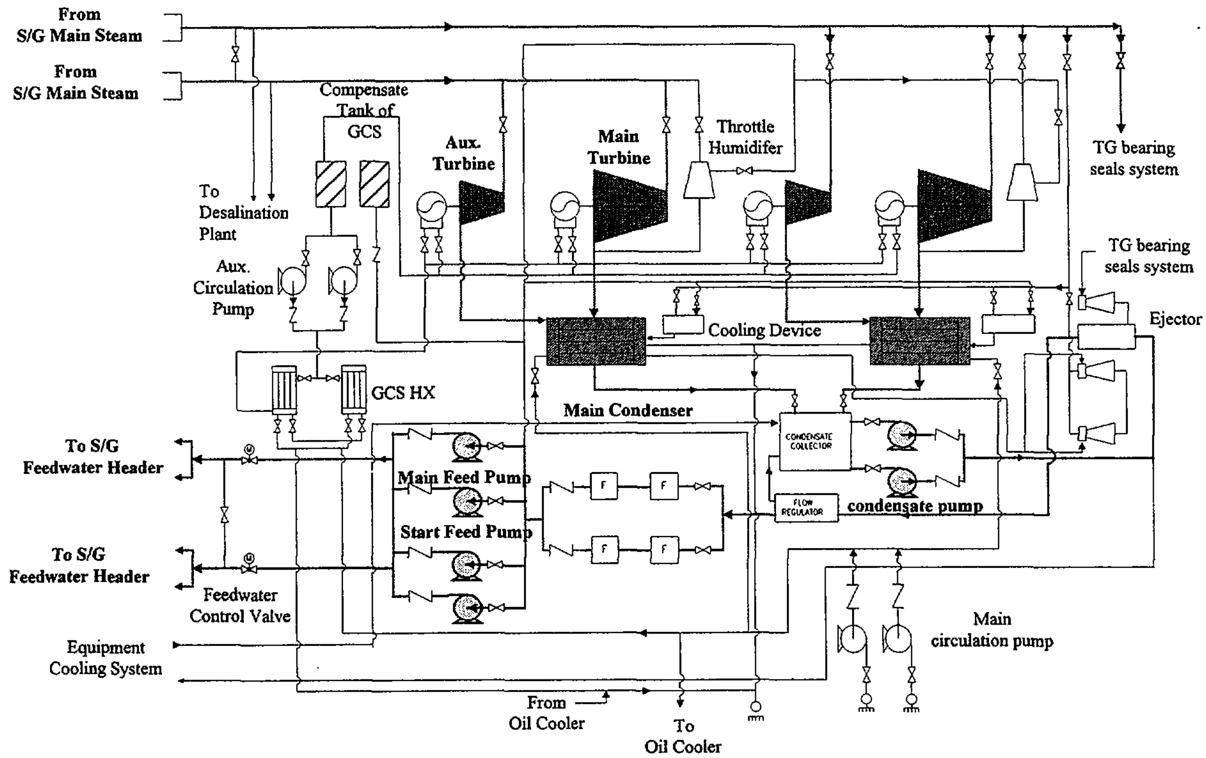

Fig. 3 SMART Secondary System

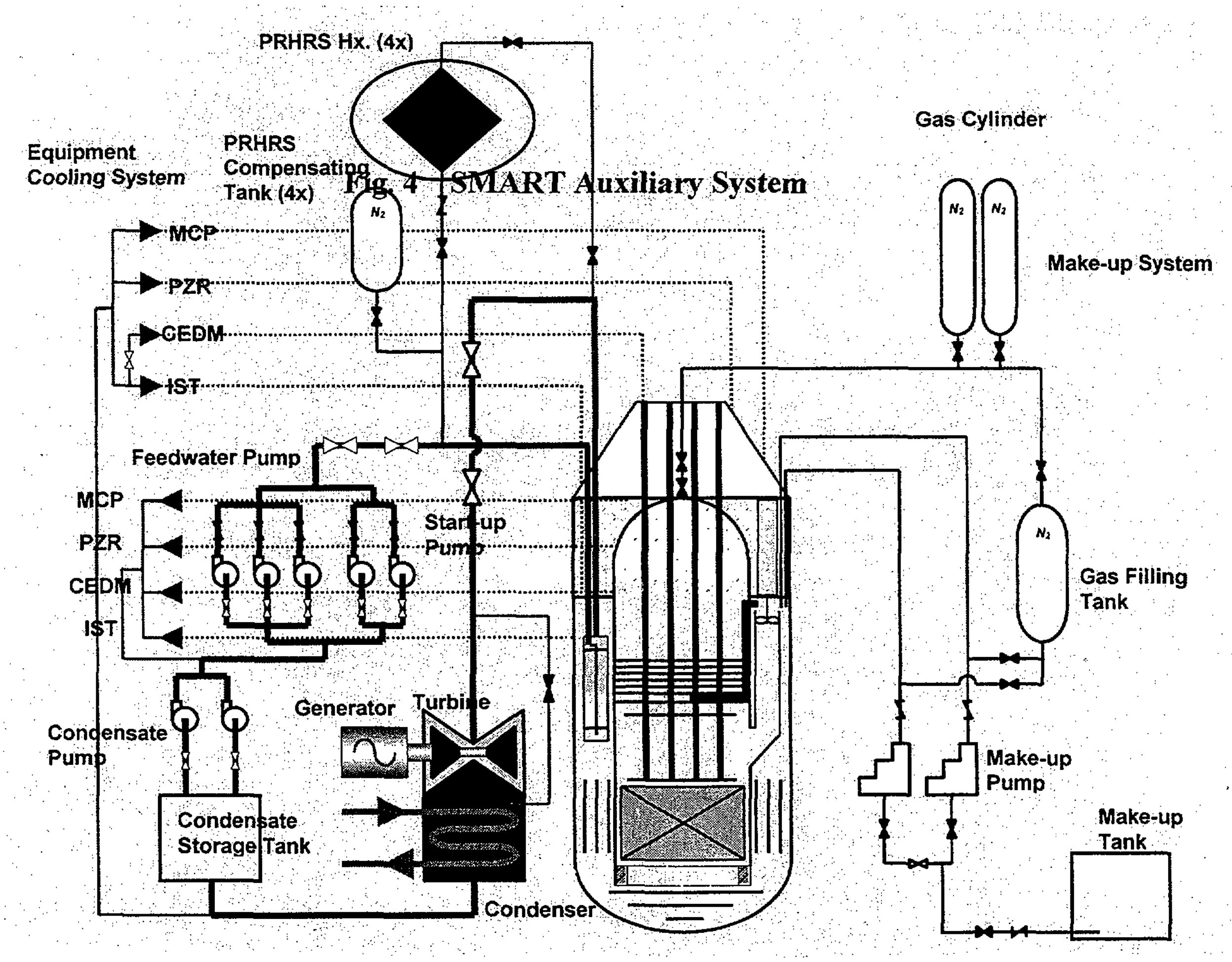

Fig. 4 SMART Auxiliary System

ADVANCED MARINE REACTOR MRX AND ITS APPLICATION FOR ELECTRICITY AND HEAT CO-GENERATION

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Abstract

The basic concept of an innovative advanced marine reactor MRX has been established by design study toward the goals of light-weightiness, compactness, and safety and reliability improvement with adoption of several new technologies. The MRX is the integral-type PWR aimed for use of ship propulsion. Adoption of a water-filled containment makes the reactor light-weighted and compact greatly. The total weight and volume of the reactor are 1600 tons and $1210 m^3$, which are equivalent to halves of the Mutsu, although the reactor power of MRX is three times greater. An engineered safety system of the MRX is a simplified passive system, function of which is confirmed by the safety analysis to be able to keep the reactor integrity even in a case of accident. Reliability of the system is evaluated by the PSA and revealed to have two orders smaller core damage occurrence frequency than existing PWRs. The MRX can be applied to an energy supply system of electricity and heat co-generation. Concept of the nuclear energy supply system is designed to generate electricity, heat and fresh water. The nuclear barge is shown to be a possible nuclear energy supply system with advantage of being movable easily.

1. INTRODUCTION

Small-scale reactors have more advantages than large-scale reactors in variety of energy utilization, construction, maintenance and adoption of innovative technologies, while the latter have scale merits in construction cost. The nuclear energy utilization of small-scale reactors are ship propulsion, electricity generation, heat supply, and sea water desalination, etc. Construction and maintenance for small-scale reactors can be made in factories exclusive use for them, but not at the site of the plant. In small-scale reactors, the safety can be enhanced sometimes by new technologies.

Japan atomic energy research institute (JAERI) have designed a small-scale integral type PWR with 100 MW of the thermal output for ship propulsion, called MRX (Marine Reactor). High priorities of design requirements are laid on being economical and reliable of the reactor system. Design goals of being light-weighted, compact, simple and safe were set up to meet the design requirements, and these goals were achieved by several new technologies such as the water-filled containment, the in-vessel control rod drive mechanisms (CRDMs). Feasibility of the MRX design concept for the commercial ship was surveyed by checking compatibility of the systems, and evaluating safety and economics in the whole system. Development of the new technologies adopted in the MRX has mostly completed for the stage of principal function confirmation. The MRX, therefore, can be realized if the economic and social circumstances allow the commercial nuclear power ships.

The MRX can be also used for a multi-purpose utilization such as electric power generation for a distance district, sea water desalination, and heat supply. Features of MRX, i.e., being easy movable, light-weighted, compact, simple and safe can be also suitable for these multi-purpose utilization.

2. MRX FOR SHIP PROPULSION

2.1. Design goals and new technologies adopted in MRX

JAERI set up the design goals for MRX with 100 MW of thermal output which is installed into the ship with 80,000 HP of the shaft power. The ship with two MRXs (100MWtx2) is to have capabilities of ice-breaking for research of the climate of north pole region, together with role of a commercial ship. This forces the MRX reactor to bear severe external and internal conditions. The concept of nuclear icebreaker with two MRXs equipped is shown in Fig. 1. The concrete design goals of the MRX are as follows:

1. Light-weightiness and compactness
   To realize the light weighted and compact reactor, the total weight including the reactor vessel, the containment and the shielding should be not greater than 1600 tons which is equivalent to nearly a half weight of the Mutsu, the first Japanese nuclear experimental ship. The total volume of the reactor system should be small as possible according to light weightiness. These goals are decided so to be competitive with the conventional engine of commercial ships from survey of needs on nuclear ship which was conducted by JAERI.
2. Simplification of the system
   The design goal for being simple is that the numbers of the sub-systems and the machinery adopted in the reactor system should be minimized and definitely less than those of other plant, for example, the conventional PWRs.
3. Safety and reliability improvement
   At least, stability in the normal operation and safety in a case of postulated accident should be confirmed by the transient analysis, and also reliability of the system by provability safety analysis.
4. Maintainability improvement
   From the economical point of view, the period for maintenance of the system or refueling of the core is required to be short as long as possible. The design goals is set up with that the period should be less than one month.

Relationship between the goals and design improvements to attain these goals are illustrated in Fig. 2. Major adoptions for design improvement are as follows.

a) The integral type reactor
b) The in-vessel type control rod drive mechanism
c) The water-filled containment
d) Passive decay heat removal system
e) One-piece removal of the reactor system

Details of the MRX design with these improvements are presented in the following chapter.

2.2. Design conditions for marine reactor

Design conditions of the MRX concerning the specific items of a marine reactor are introduced as follows.

a) Even at the low temperature of 300K, the reactor should be able to be shut down without use of a soluble poison by taking account of ship sinking. That is, the control rods should have ability to maintain the reactivity K_{eff} less than 0.99.
b) The reactor should have ability to generate over 30% of the rated output for steerageway (ship propelling with effective rudder), under condition of the one rod stuck.
c) The reactor should respond to a rapid and/or large shipload change such as turbine trip or ahead and astern maneuvering. The reactor should slave to the change from the base load (15% of the

rated) to the rated load in 30 seconds. These design conditions are used for designs of the fuel and the reactor automatic control system.

d) The reactor should bear the external forces due to ship motions. The external forces to be considered are that the maximum allowable vertical and horizontal acceleration, rolling angle and heel angle are $\pm 1g$, 45 degrees and 30 degrees, respectively. These conditions for the thermal hydraulics design of the core are decided by considering the ship navigation in the North Pole Sea region.

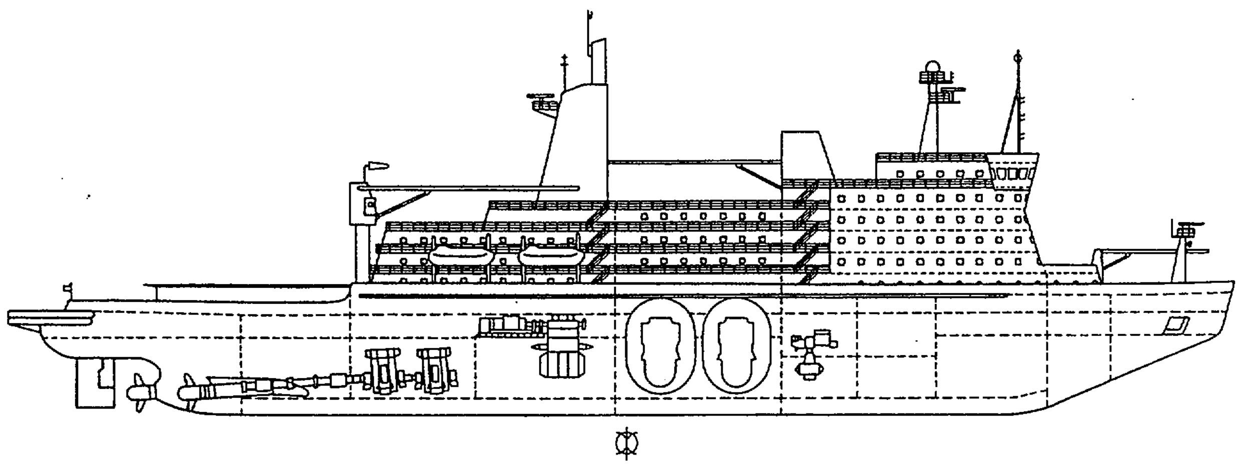


Fig. 1 Two MRX equipped icebreaker for scientific observation

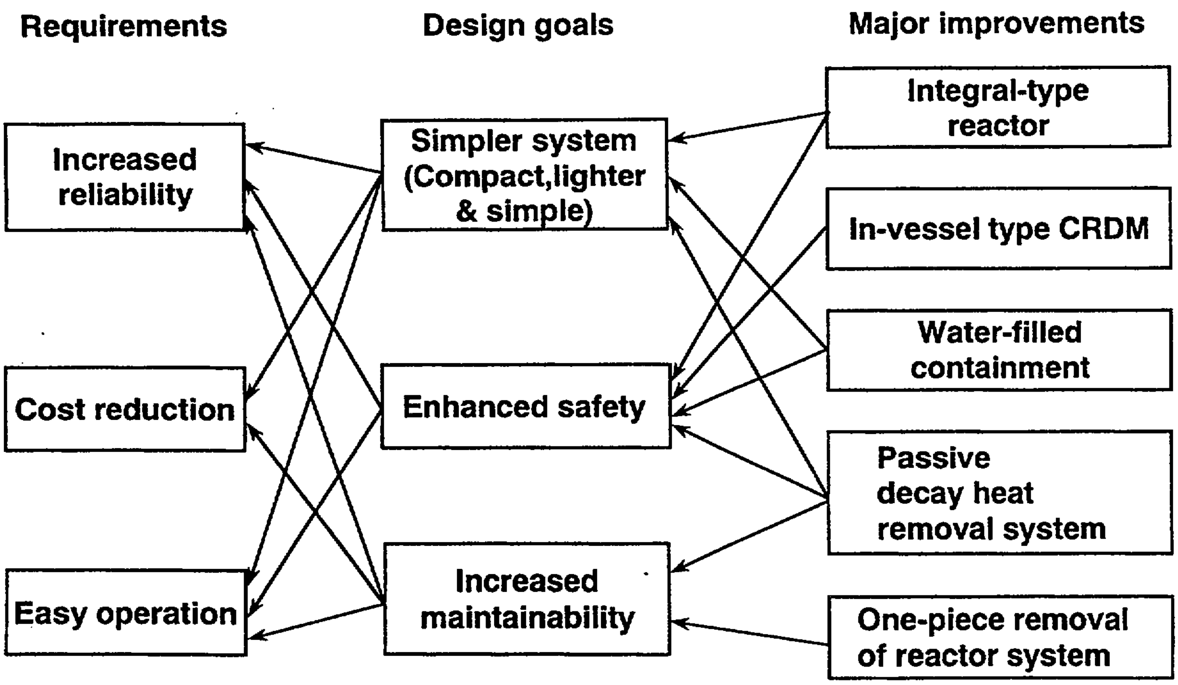


Fig. 2 Design goal and major improvement of MRX

2.3. Reactor system design

(1) Integral type reactor

A cross-section of the reactor pressure vessel together with that of the containment vessel is shown in Fig. 3. Effective layout of the primary components makes the reactor compact by installing the most of components inside the vessel: The core locates in the lower part, the steam generator in the middle part, the CRDMs and pressurizer in the upper part inside of the reactor pressure vessel, and the primary coolant pumps are connected directly to the flange of this vessel. Major specifications of the core, the CRDMs, the SGs, the pressurizer, and the primary coolant pumps are shown in Table 1.

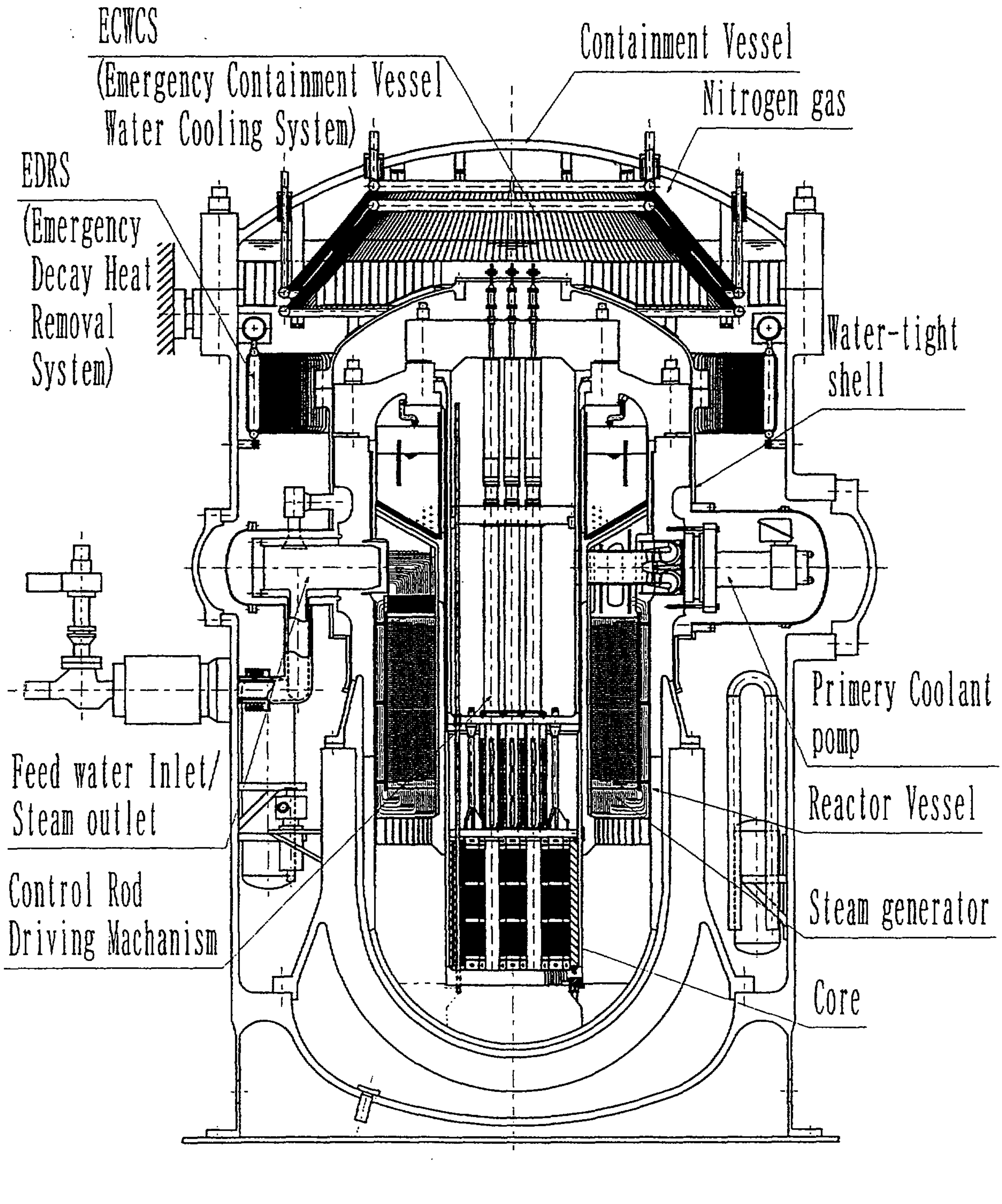


Fig. 3 Conceptual drawing of MRX

Table 1 Major parameters of MRX

Reactor Power	100 MWt	Main coolant pump	
Reactor type	Integral type	Type	Horizontal axial flow canned motor type
Reactor coolant		Rated power	200kW
Operating Pressure	12 MPa	No. of pumps	2
Inlet/Outlet Temp.	282.5 / 297.5 °C	Steam generator	
Flow rate	4,500 t/h	Type	Once-through helical coil type
Core / Fuel		Tube material	Incoloy 800
Equivalent Dia	1.49 m	Tube outer/inner dia.	19 / 14.8 mm
Effective height	1.40 m	Steam temp. /press.	289 °C / 4.0 MPa
Ave. linear heat flux	7.9 kW/m	Steam flow rate	168 ton/h
Fuel type	Zry-clad UO_2 fuel	Heat transfer area	754 m^2
U-235 enrichment	4.3 %	Reactor vessel	
Fuel inventory	6.3 ton	Inner dia. / height	3.7 / 9.7 m
Fuel Ave. burn-up	22.6 GWd/t	Containment	
No. of fuel assembly	19	Type	Water-filled RV immersion type
Fuel rod outer dia.	9.5 mm	Inner dia. / height	7.3 / 13 m
Control rod drive mechanism		Design press.	4 MPa
Type	In-vessel type		
No. of CRDM	13		

Steam generators (SGs) of the MRX are of the once-through, helical coil tube type being suitable to the integral type reactor, which were adopted in the Otto-Hahn. The primary cooling water flows outside of the tubes, and the secondary water and steam flow inside of the tubes. The SGs are hung from the main flange of the reactor pressure vessel (RPV). To simplify handling in the in-service inspection and the refueling, the SGs are not removed from the RPV. The tubes of SGs can be inspected by inserting an inspection prove from the headers.

The pressurizer of ringed flatness type is placed outside the CRDM to use the space effectively in the RPV. The pressurizer comprises the heater, the spray, the relief valves, and the safety valves. The set-point of the water level is chosen so to prevent the heater from uncovering in the water and the spray injection holes from flooding even in the cases of 30 degrees of ship inclination and rapid load change. Water inside the pressurizer passes to the hot leg through surge holes at the bottom of it.

The primary coolant pumps of a canned motor type, the double piped suction and delivery model is set horizontally at the upper part of the SGs. The JRR-2 (Japan research reactor No.2) and the Lenin had the experiences on the pump of horizontal canned motor type which revealed to have two problems;

- difficulty in selecting the material and the structure for the radial bearing to endure a non-uniform load of turbine rotor support, and
- deformation of the motor due to a non-uniform temperature profile when the pump stopped in the state of high temperature.

The counter-measures of the MRX design are employment of carbon radial bearing to the former problem, and employment of slim motor and contrivance for uniform cooling of motor to the latter one.

The primary system pipes connected to the RPV are of small sized diameters, and the largest one is 50mm of diameter for those of the safety valve and the emergency decay heat removal system. A large LOCA over 50mm diameter, therefore, is not necessary to be postulated. The elevations of the all pipes connected to the RPV are near the primary coolant pumps as a result of possible upper part of the RPV.

(2) Reactor core

Major parameters of the MRX core are shown in Table 2. Since the MRX adopts the control rods for power control and eliminates the chemical shim to avoid re-criticality due to sea water entering into the core in a case of ship sunk, parameters on the reactor physics are characterized by rather large negative moderator density reactivity coefficient (α_m) for the core cycle life. The large value of α_m can contribute effectively to self-regulating reactor control property even for a heavy load change as described in the reference [1]. On the other hand, each control rod cluster has a large value of control reactivity which is a disadvantage for a possible rod ejection accident (REA). For the REA, however, adoption of the in-vessel type CRDM can eliminate the possibility of accident.

The core power profile of the MRX is not as uniform as the core adopting the chemical shim. To flatter the power profile, the MRX adopts the fuel rods with Gd_2O_3 and the burnable poison rods (BPs) filled up with boron glass, and increases the number of control rods per the fuel assembly. The nuclear characteristics were evaluated on the basis of core analyses with the SRAC[2] code system and the Monte Carlo code, MVP[3]. Although the total peaking factor of core power distribution is relatively large (3.98), the maximum of linear heat rate is 30.4kW/m which has enough margin to the limiting value 41kW/m for fuel rod designing. Two-batch fuel shuffling strategy is adopted to obtain high burn-up without increase in excess reactivity at the beginning of core cycle, in addition to adoption of the fuel rods and BPs mentioned above. Refueling will be able to be performed on a dockyard or land based facility on the same period of the mandatory hull survey per four years. The average burn-up is 23GWD/t and the life of core cycle is eight years, by assuming the core load factor of 50 % and satisfying the design conditions of a) and b) described in the sec. 2.2.

For thermal hydraulic parameter, the minimum DNBR is evaluated by taking account of effect of ship motion, satisfying the design conditions of (iv). The critical heat flux (CHF) under condition of heaving is known from some experiments to become small. There is not yet the experiment on the CHF covering the MRX operation condition. Ishiki's correlation[4] on the CHF based on the experiments by the conditions of atmospheric pressure and natural circulation flow is applied to MRX evaluation by the conservative way, and the 25% of CHF reduction due to heaving (± 1g) applied for DNBR evaluation. The Minimum DNBR for the rated operation 2.25 as shown in the table is evaluated using COBRA-IV code with EPRI's DNB correlation. Since the acceptable minimum DNBR is 1.73 (=1.3x (1/ (1-0.25), where 1.3 is an uncertainty factor the same as the W-3 correlation), the value of 2.25 for the rated operation is shown to have enough margin to the acceptable limit.

(3) In-vessel type control rod drive mechanism

The whole structure of the CRDM is shown in Fig. 4. The CRDM consists of a driving motor, a latch magnet, separator ball nuts, a driving shaft, a rod positioning detector, a spring and other components. These CRDMs work in the primary loop water of which condition is very severe, the high temperature and pressure (583K,12MPa). The CRDM is under development at the JAERI since such CRDM that works in the severe condition does not yet exist. Details of the CRDM are described in the reference[5]. Specific functions required to the CRDMs used for marine reactor are to be able to scram the reactor within a given time limit and afterward to maintain the shut-down state even at conditions of ship inclination or heaving. To suit this, a strong scram-spring is adopted in these CRDMs. For example, in a case of 90 degrees of ship inclination, the CRDM can be inserted completely in the core within five seconds (in the normal state, within 1.4 second). And the shut-down state can be maintained by the strong spring.

(4) Water-filled containment

Functions of the water-filled containment are to passively maintain core flooding in cases of accidents including a LOCA, to shield radiation, as well as to enclose the area for prevention of radioactive materials release to a surrounding. The main cooling system with containment water is shown in Fig. 5. There is a nitrogen gas in the upper space over the water surface of the containment. Core flooding can be maintained passively by pressure balance of the containment and

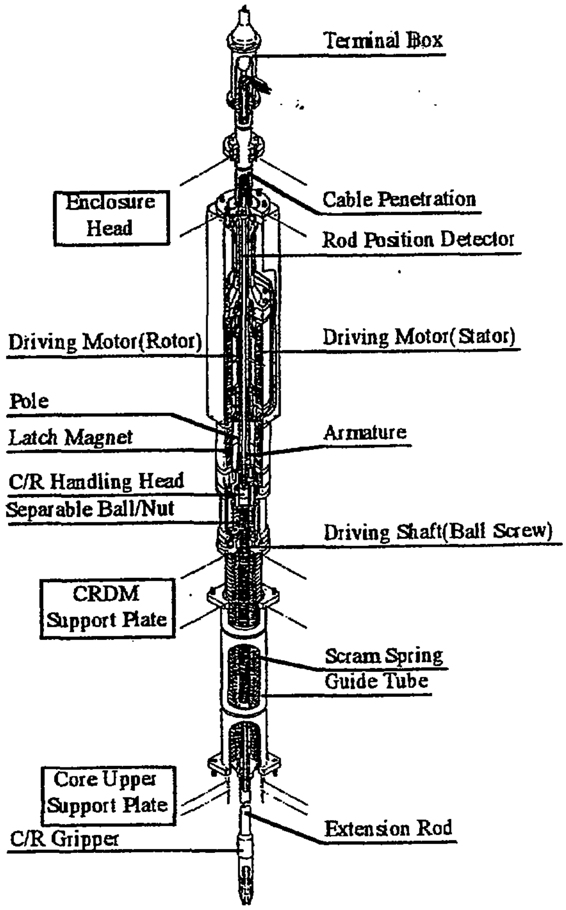
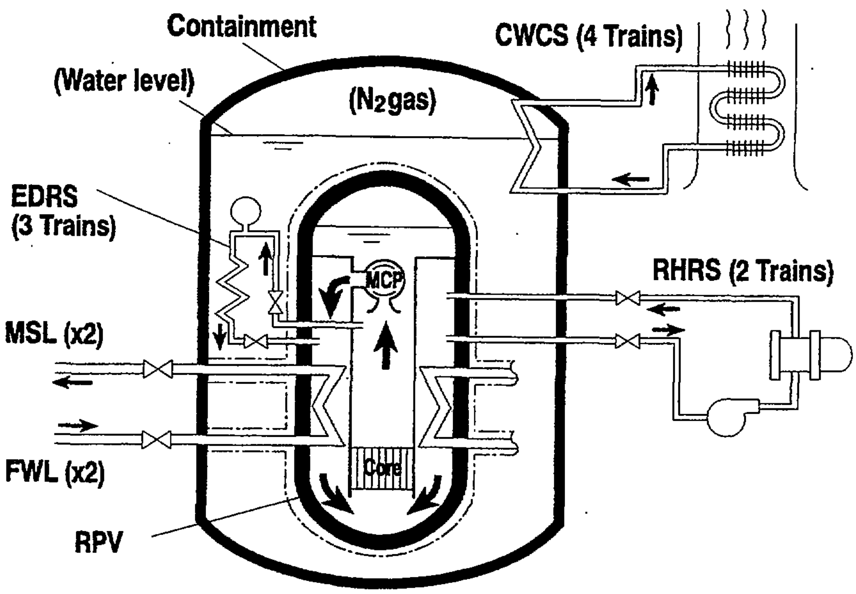


Fig. 4 Concept of an in-vessel type CRDM



EDRS : Emergency Decay Heat Removal System
CWCS : Containment Water Cooling System
RHRS : Residual Heat Removal System
MSL : Main Steam Line
FWL : Feed Water Line

Fig. 5 Reactor cooling system with decay heat removal system

RPV, and with helps of EDRS and CWCS. Details will be described in later (in LOCA analysis). Thus, the core flooding in a LOCA can be attained passively without ECC pumps or an accumulator if the containment initial water level is appropriate. Experiment [6] on the thermal hydraulic behaviour of water-filled containment was conducted and the relationships between the initial water level of containment and the balance pressure, etc., were obtained, and the principal function of core flooding was confirmed experimentally.

Table 2 Major parameters of MRX core

Parameters		Design Condition
Reactor physics parameters		
k-eff		
BOC (cold shutdown, all control rod clusters are inserted)	0.82962	
EOC (hot and full power, all control rod clusters are withdrawn)	1.02041	> 1.02
Reactivity coefficient		
BOC Doppler coefficient	-2.2×10^{-5} Δk/k°C	
Void coefficient	-2.5×10^{-3} Δk/k/% void	
Moderator density coefficient	3.0×10^{-1} Δk/k/(g/cm^3)	
EOC Doppler coefficient	-2.3×10^{-5} Δk/k°C	
Void coefficient	-2.6×10^{-3} Δk/k/% void	
Moderator density coefficient	3.2×10^{-1} Δk/k/(g/cm^3)	
Reactivity shutdown margin	2.17 % Δk/k	> 1.0 %
Thermal hydraulic parameters		
Heat flux	1.427 kW/m^2	
Maximum linear heat rate	30.4 kW/m	< 41 kW/m
Fuel center temperature (full power, 1,200 MWD/t)	1,785 °C	
Minimum DNB (100 % power)	2.25	> 1.73

Inside the containment, the RPV, the primary loop piping, the pressurizer spray pipe, the EDRS, the CWCS etc., are installed in the water as shown in Fig. 3. A thermal insulation, therefore, is necessary to prevent the heat loss from the surface of these components into the water. Thermal insulation structures are arranged except the heat exchangers of the EDRS and the CWCS. Especially, the RPV is covered with a water-tight shell made of stainless steel with 45mm of the thickness. Between the RPV and the water-tight shell, insulation of stainless steel felt is inserted. The heat loss from the RPV with this insulation is estimated less than 1% of the rated power. Leakage into the space between the RPV and the water-tight shell can be detected by a moisture detector of nitrogen gas circulating inside the space.

Water inside the containment has also a role of radiation shielding, which can eliminate the concrete shield outside the containment. This merit makes the MRX plant drastically light weighted. Radiation shielding performance was evaluated with the use of the discrete ordinates codes the ANISN[7] and the DOT3.5[8], and the point kernel code QAD-CGGP2[9]. The dose rates equivalent are sufficiently small compared with the design criteria shown in Table 3 for the rated operation, the reactor shut-down, and the hypothetical accident. Details of the analysis are described in the reference[10]. For shielding, besides the water inside the containment, a steel shield is inserted between the core and the steam generator and a cast steel shield of 45cm of thickness is set outside the RPV, together with making the core barrel thick.

(5) Passive decay heat removal system

In the normal operation manual, the reactor decay heat after the reactor shut down is to be removed through the steam generator and the residual heat removal system like the land based PWRs. When this normal procedure is not available due to an accident, the decay heat is removed passively by helps of the EDRS and the CWCS as shown in Fig. 5. The decay heat is transferred from the primary coolant to the water of the containment through the heat exchanger of EDRS, and

Table 3 Design parameters for dose rate equivalent

Area	Design criteria	Access frequency
Reactor room	≦10 μ Sv/h	48 hours per week
Inside double bottom	≦500 μ Sv/h	1 hour per week
Between below containment vessel and double bottom shell	≦5 mSv/h	not necessary to access in ordinary condition
Engine room	≦6 μ Sv/h	Surveillance area
Accommodation Area	≦0. 0057 μ Sv/h	Outside surveillance area boundary ≦50 μ Sv/year
Outside ship side	≦0. 11 μ Sv/h	Above waterline Outside surveillance area boundary

from the water of the containment to the atmosphere through that of CWCS. The flow is driven by only natural circulation force. The EDRS begins to transfer the heat by opening the valves, but the CWCS of heat-pipe type transfers always the heat because of no valve setting. For this procedure, only opening of the EDRS valve is needed, and a small power source is enough to open the valves.

The EDRS and the CWCS have function as the engineered safety system, and are designed by taking a single failure criteria. That is, the EDRS has three trains with each 100% capacities, by taking account of one line broken as initiation of the LOCA. The CWCS has four trains with capacity of 100% per three trains, 33% of one train.

(6) One-piece removal of reactor system

Maintenance and refueling of the reactor system is made by the one-piece removal of reactor system. The concept of the method is shown in Fig. 6. The containment with the internals including the PRV and the core is removed from the ship at a dockyard which has the facility of exclusive use for maintenance. After the removal, a new containment of which maintenance is already completed can be replaced to shorten the maintenance period.

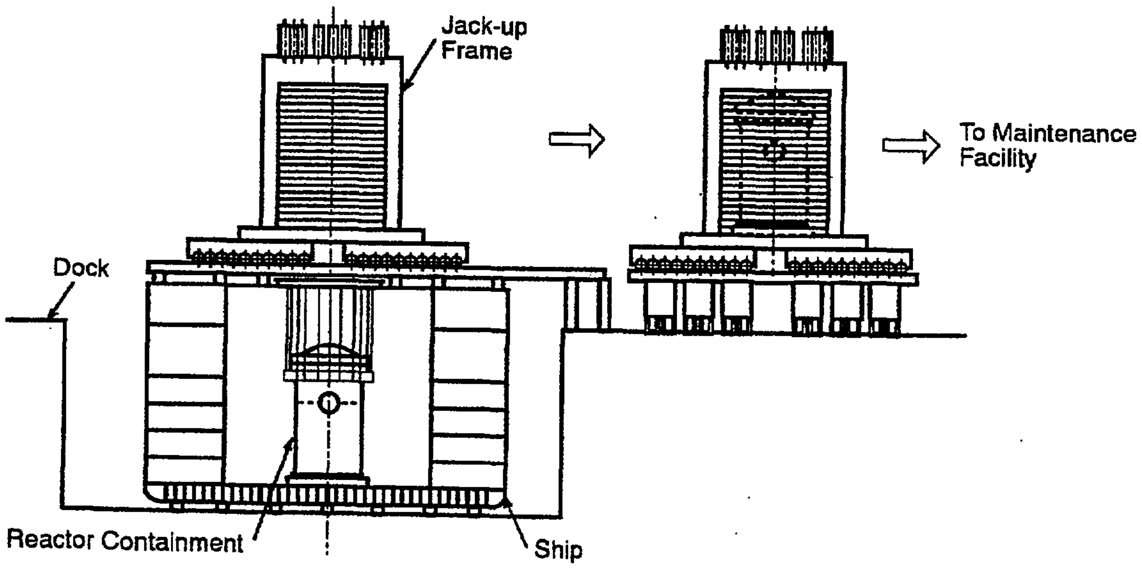


Fig. 6 One-piece removal system

2.4. Characteristics of transient response

As described in the previous chapter, the MRX copes with anomaly including accidents by help of the engineered passive safety system: The core is flooded by the water-filled containment and the decay heat is removed by the EDRS and the CWCS. To view the function and the transient behavior, the LOCA analysis using RELAP5/mod2[11] and COBRA-IV[12] codes is presented here as followings.

Rupture of the EDRS pipe of 50mm diameter is presumed to initiate the LOCA. The transient is shown in Fig. 7. The reactor pressure decreases rapidly immediately after rupture, and it does slowly from about 40 seconds to 520 seconds due to change of break flow phase from the water to the steam. Reactor scram is initiated by the signal of reactor pressure low level at 100 seconds. The EDRS of an intact loop starts to operate at 300 seconds.

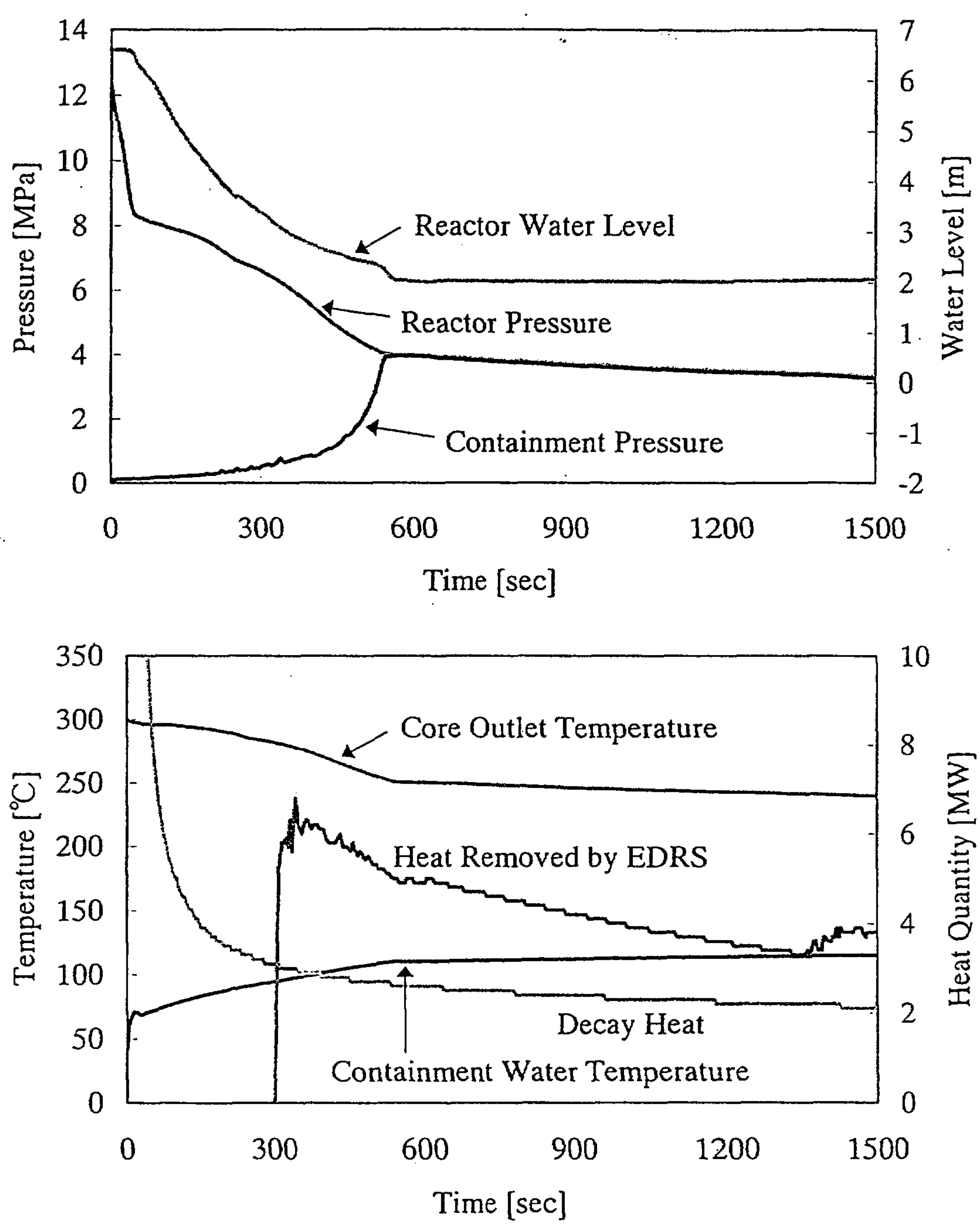


Fig. 7 LOCA analysis for break of 50 mm diameter pipe

The containment pressure rises due to inflow of the primary coolant through the broken pipe. Both of the containment pressure and the reactor pressure equal at 520 seconds, and afterwards they show almost a constant value. The containment pressure is under the design limit throughout the transient.

The water level of the core decreases due to outflow of the primary coolant. It, however, remains at about 2m high level over the top of the core when both of the pressure equals. The water level 2m is enough for core flooding completely even if the ship inclines at 30 degrees, which is one of the design conditions mentioned previously.

The decay heat is removed after the EDRS valve open as follows. The upper part of RPV is covered with the steam at this time. This steam flows into the EDRS, and be cooled to the state of water through passing the heat exchanger. The water returns to the down comer of the RPV. Thus a natural circulation loop is formed by the EDRS and the RPV, and the decay heat is transferred to the water of containment. In the containment water, the heat is transferred by single-phase natural circulation from the EDRS to the CWCS. The CWCS transfers this heat to the atmosphere. After 300 seconds, the heat removal rate is larger than the decay heat rate and the temperature of core outlet decreases gradually. After 600 seconds, the heats removed by the CWCS and the EDRS balance, and the temperature of containment water remains to a constant.

During the transient, the fuel cladding temperatures are not higher than the initial value of the steady state and the core is flooded, so that the minimum DNBR never falls down from the initial value. This leads to a greater contribution for demonstration of MRX's outstanding feature of safety. As a result, core flooding and decay heat removal are revealed to be done appropriately by the engineered passive safety system.

On the other hand, in the normal operation, frequent severe load changes will induce the MRX. The analysis by RELAP5 shows that the MRX can operate stably for these load changes without calling the scram or relief valve operation of the pressurizer, though the results are not shown here.

2.5. Achievement of design goals

(1) Light-weightiness and compactness

Breakdown of weights and volumes of the reactor plant components is shown in Table 4. The total weight and the volume of the MRX are approximately 1600 tons and 1210m^3, respectively. The total weight is summation of all the structures in the containment vessel, the filled water and the containment vessel itself. The weight is not greater than that set for the design goal, that is, light-weightiness is achieved. According the light-weightiness, the volume is also small as shown below.

Comparison with the MRX and Mutsu or the three trial designs[13] studied by JAERI is shown in Fig. 8. The MRX has a half weight of the Mutsu even though the reactor power of MRX is three times greater than that of Mutsu, and also a half weight of the trial designs. Because those other than the MRX have the secondary shields, they are heavier than the MRX. The volume of the MRX is the half of that of Mutsu and 40% of that of the trial design studies.

Contributions to light-weightiness and compactness of the MRX are due to adoption of integral type reactor installing the steam generator and the pressurizer inside the RPV, and adoption of water filled reactor containment vessel. The reactor room and reactor component room are not subject to the goal in the present discussion, but the total volume of them in MRX is 3326m^3, which is almost the same volume as that of the Mutsu. Thus, the MRX has also the compact reactor room and reactor component room, since the engineered safety system is greatly simplified, and the chemical shim for the reactivity control is not adopted.

(2) Simplification of the system

The systems of the MRX for the normal operation are almost the same as existing PWRs except the chemical processing system. Elimination of the chemical processing system makes the normal operation system of the MRX be simple greatly.

Table 4 Breakdown of weight and volume of MRX reactor component

Component	Weight (ton)
Reactor vessel	280
Steam generator	57
Support structure	7
Pressurizer	7
Primary coolant pump	11.4
Containment	550
Emergency decay heat removable system	3.6
Containment water cooling system	10.3
Reactor core internal structure	
• Upper core structure	18.6
(Inner tank, control rod drive mechanism support plate, ... etc.)	
• Lower core structure	21.3
(Core tank, neutron relfector, lower core plate, ... etc.)	
• Control rod drive mechanism	7
(13 drives, control rods, drive axes, acceleration mechanism, guide tubes, ... etc.)	
• Fuel assemblies	9.9
Primary coolant	45
Containment water	238.6
Valves	0.5
Shields	235
Auxiliary system and tubes	50
Total	1552.1

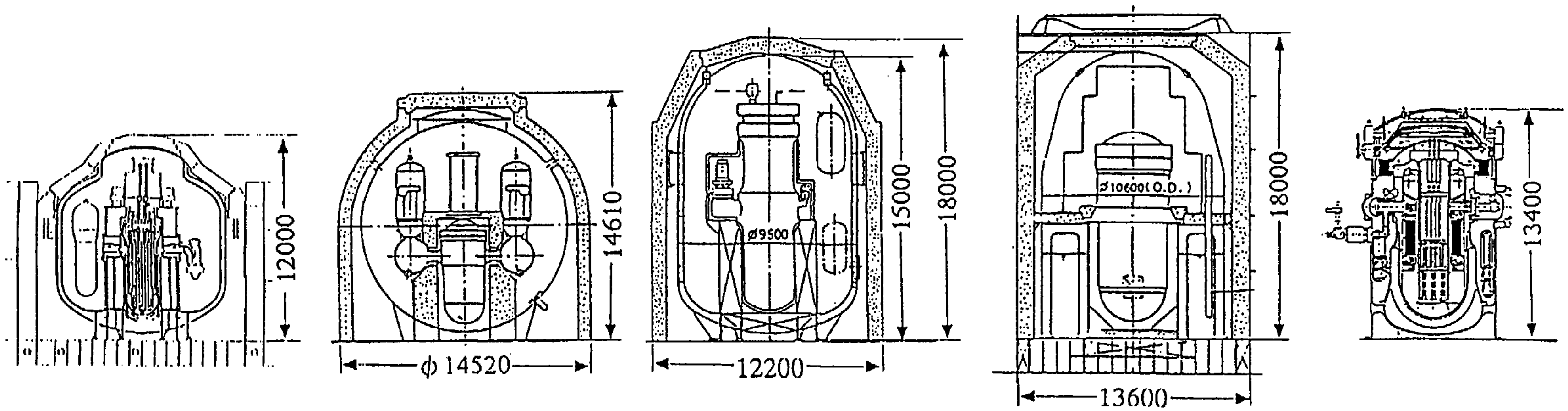
The engineered safety system of the MRX is also simplified significantly. Comparison of the engineered safety system with the MRX and the Mutsu, existing PWRs, and the next generation reactor AP-600[14] is shown in Table 5. To maintain the core flooding in an accident, the Mutsu and existing PWRs have the injection systems or the accumulator actuated by the active components such as pumps and valves. On the other hand, the MRX has only water in the containment vessel. To remove the decay heat in a case of anomaly, the Mutsu and existing PWRs have the cooling system including the heat exchanger and the pumps while the AP-600 has cooling systems including the heat exchanger and the passive operating loop. The MRX adopts the similar passive operating loop to the AP-600. The AP-600 and the MRX use a passive power source, the battery because only small power is enough for operation.

The numbers of constituent equipment for the engineered safety system are compared in Table 6. For core flooding, the Mutsu and the existing PWRs have approximately 95 equipment while the AP-600 does 45 equipment. The MRX does not have any active equipment to keep core flooding. For the decay heat removable system, both of the AP-600 and the MRX have small numbers of equipment while the Mutsu and the existing PWRs have large numbers of equipment. Among these plants, the MRX has the smallest number of constituent equipment.

This allows to say that the goal of system simplification of the MRX is achieved. The simplification of the systems leads to improvement of economy and reliability through the reductions of plant construction and maintenance cost, the human error during maintenance, and the probabilities of equipment failure.

(3) Safety and reliability improvement

The functions of the engineered passive safety systems of the MRX which are significantly simplified are evaluated with the safety analyses. The LOCA analysis as an example presented in the previous chapter shows that core flooding is kept and decay heat removal is performed successively. Corresponding to the Japanese governmental "PWR licensing safety review guideline", the analyses of the accidents and the anticipated transient events are conducted. The



	Mutsu reactor	Semi-integral type reactor	Integral type reactor	Integral type reactor with the self-pressurized pressurizer	MRX
Weight	about 3200t	about 2600t	about 2900t	about 3200t	about 1600t
Installation space volume (involved secondary shield)	about 2300m^3	about 3080m^3	about 2700m^3	about 3300m^3	about 1210m^3
Reactor room + Component room volume	3149m^3				3326m^3

Fig. 8 Comparison of reactor weight and volume

Table 5 Comparison of engineered safety system with other nuclear plants

	PWR(2loop)	Mutsu PWR (2 loop)	AP600	MRX
Core make-up & keep core flooding	High pressure coolant injection system (2) Low pressure coolant injection system (2) Accumulator tank (2)	High pressure coolant injection system (2) Low pressure coolant injection system (2)	Refueling tank (1) Core makeup water tank (2) Automatic depressurization system (2) Accumulator tank (2)	Containment vessel water (Water-filled containment vessel) (1)
Long term coolant	Low pressure coolant injection system (2)	Re-circulation coolant system (1)	Static residual heat removal system (2) (Natural circulates)	Emergency decay heat removal system (3) (Natural circulates)
Containment vessel coolant	Containment vessel spray system (2)	Containment vessel spray system (1)	Static containment vessel coolant system (1)	Containment vessel water coolant system (4)
Emergency power	DC power (1) Emergency generator (2)	DC power (1) Aux. generator for Emergency (2) Emergency generator (1)	DC power	DC power (2)

( ): Nos. of system

Table 6 Comparison of components of engineered safety system

Component	PWR(*) (2 loop plant)	Mutsu PWR (2 loop plant)	AP600(*)	MRX
(Safety Injection system)				
• Pump	4	7	0	0
• Tank	45	2	5	0
• Heat exchanger	0	1	0	0
• Remote operation valve	45	85	40	0
(Residual heat removal system)				
• Pump	5	5	0	0
• Heat exchanger	2	2	2	3
• Tank	1	2	0	0
• Remote operation valve	19	15	4	3
(Safety heat sink system)				
• Pimp	6	4	0	0
• Tank	2	1	1	0
• Mechanical draft cooling tower	4	2	0	0
• Heat pipe type heat exchanger	0	0	0	4
• Remote valve	34	18	4	0
(Containment spray system)				
• Pump	2	2	0	0
• Tank	1	0	0	0
•Remote operation valve	8	2	0	0

(*) IAEA, "Review of design approaches of advanced pressurized LWRs", IAEA-TECDOC-861, 1994

former includes the LOCA, the SGTR, the main steam line break and the feed water line break, and the latter does the loss of the primary coolant flow and so on. The control rods ejection accident appeared in the guideline is neglected for the MRX since the MRX adopts the in-vessel type CRDMs which eliminate the possibility of this accident. Typical results of the minimum DNBR, and the maximum reactor pressure, $P_{max,RV}$ are summarized in Table 7. The results show to satisfy the standards referring criteria in the PWR licensing safety review guideline.

A countermeasure for the SGTR of this plant is important because the water in the containment is not useful for the core flooding by the water-filled containment. Isolation of the steam generators is a critical function to halt flowing out of the primary coolant through the steam generators. The isolation valves of the MRX are to be operated by a signal of reactor pressure low

Table 7 Result of safety evaluation on accidents

Matter	Scram signal	Reactor pressure (MPa)	Minimum DNBR
LOCA	Reactor pressure low	12.2	2.29
Loss of primary coolant flow accident	Power range neutron flux high	13.8	2.07
Main feed water line break accident	Reactor pressure high	13.8	2.15
Main steam line break accident	Power range neutron flux high	13.8	2.31
SG heat transfer pipe rapture accident	Reactor pressure low	12.2	2.09

Where,
Maximum allowable reactor pressure = 16.4 MPa,
Minimum allowable DNBR = 1.73.

and isolated to terminate the flow. The decay heat can be removed by EDRS even if the SGs are isolated. Therefore, reliability enhancement of the isolation is especially important for this plant to keep the core flooding. To enhance the reliability of the isolation, redundancy and diversity of the valves are realized by adopting air operating valves and electric motor driven valves arranged in series.

Reliability of the reactor system is evaluated on the basis of the probabilistic safety analyses (PSA) by the event tree method on a LOCA, a SGTR, and others. As shown in Table 8, the total occurrence frequency of core damage is approximately 3.5×10^{-8} /reactor/year. Contribution of redundancy and diversity of the isolation valves to enhancement of reliability is confirmed by the PSA. Details of the analyses can be found in the reference[15]. The total occurrence frequency of the core damage of the MRX is two orders of magnitude lower than that of existing PWR plants such as Surry, Sequoyah, and Zion.

According to the above discussions, it can be said that safety of the MRX are ensured and reliability of the plant is also significantly improved.

(4) Maintainability improvement

By adoption of the one-piece removal of the reactor system, the ships are required to stay in the dockyard at only about three weeks for maintenance or refueling work. This method is considered to be promising since the containment with the internals is relatively small and light-

Table 8 Core damage frequency for major initiating events

Initiating Events	Core damage frequency(*)
Steam generator: heat tube rupture	7.2×10^{-9}
Volume control system: pipe rupture in containment vessel	1.7×10^{-9}
Trip of 2 main power generators	1.0×10^{-8}
Transient events	3.4×10^{-9}
Main steam system: common pipe header rapture	1.5×10^{-9}
Decay heat removable system: failed open of isolation valve	1.1×10^{-8}
Pipe rupture of sampling system	7.5×10^{-10}

(*) The values are revised ones from the previous analysis [16]. Especially the value of "pipe rupture of sampling system" is greatly lowered by reevaluating pipe rupture frequency taking account of the pipe length inside the containment vessel.

weight. Furthermore, this method can provide high quality assurance of the work in a land facilities which of space will be wide enough. The decommissioning of the reactor system can be made easily. Maintainability of the MRX, therefore, can be said to be improved.

2.6. Evaluation of Economy

For evaluation of economy, the costs of construction and whole operation of ship should be taken into account. In cost evaluation, the units of TEU and RFR are used: The TEU is a cargo capacity of container unit, Twenty-feet Equivalent Unit. The RFR is an operation cost to transport one container during the ship service life of 20 years, Required Freight Rate, $/TEU.

Comparison of the RFR between the nuclear power and diesel engine are shown in Fig. 9 for the container ships of 6,000 TEU and 30 knots of ship speed. The nuclear ship equips with two MRXs, which of the total power is increased to 348MWt. The navigation route is taken as Asia - North America. The commission is taken as 20 years from 2015. The crude oil price is assumed to be $36/bbl. The comparison shows that (a) the capital cost of the nuclear ship is about two times larger than that of the diesel ship, on the other hand (b) the fuel cost of the nuclear ship is about half of the diesel ship. In this evaluation, the environmental cost of diesel engine accounts for 22 % of the RFR. With parameter of ship speed, the comparison of the RFR is shown in Fig. 10. The figure shows that the nuclear ship has more advantage with increase of ship speed. Over 28 knots of ship speed, the nuclear ship becomes more economical than the diesel engine ship, since the weight of environmental cost becomes larger for the diesel engine ship.

3. APPLICATION OF MRX FOR SUPPLY OF ELECTRICITY AND HEAT

Besides the ship propulsion, the MRX can be applied for energy supply to ground utilization. Conceivable variable utilization of nuclear energy from the MRX are as follows:

- electricity generation;
- electricity generation and sea water desalination;
- electricity and hot water (or heat for air ventilation) co-generation;
- electricity and hot water co-generation and sea water desalination.

In the variable utilization, the nuclear plant with MRX has merits such as being easy to move, independent of strong construction site, easy to operate, safe, reliable, and so on. With these merits, a barge type nuclear plant or an offshore floating nuclear plant can supply the energy for small city or remote region where the infrastructure is not provided. Images of a barge type nuclear plant or an offshore floating nuclear plant with the MRX are shown in Fig. 11. Rescue vessel in a case of disaster caused by e.g. earthquake can be one of conceivable applications.

The energy of 300MWt is a possible output to supply electricity, fresh water and hot water for the population of 100,000 person. In this chapter, the reactor output for the energy supply plant is taken as 300MWt. Although the thermal output of the original MRX design is 100 MW, power–up of MRX is possible without change of the design concept. Three sets of the MRX of 100MWt output is another option, which has merit of redundancy in the power source and flexibility in power variation of output.

3.1. Nuclear Energy Supply System

Concept of nuclear energy supply system (NESS) is shown in Fig. 12. The NESS consists of electricity generation system of turbine and generator, heat supply system, desalination system of the multi-stage-flash (MSF) and the reverse osmosis (RO), together with the reactor. Main steam of 4MPa from the SG works at the high-pressure turbine. Exhausted steam of 150 C at the outlet of the high-pressure turbine can be used for heat supply partially and for the MSF distillation plant. The power for the RO plant can be supplied from the electricity generated. Heat transport to heat supply, as well as the fresh water generation, is provided using three-circuit flow scheme with a pressure barrier between circuits.

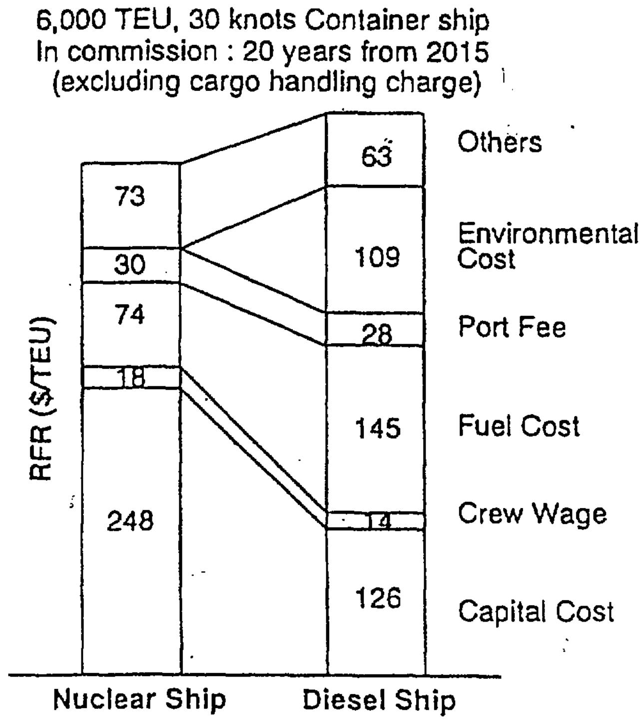


Fig. 9 Comparison of RFR between nuclear power and diesel engine for container ship with 6,000 TEU at speed of 30 knots

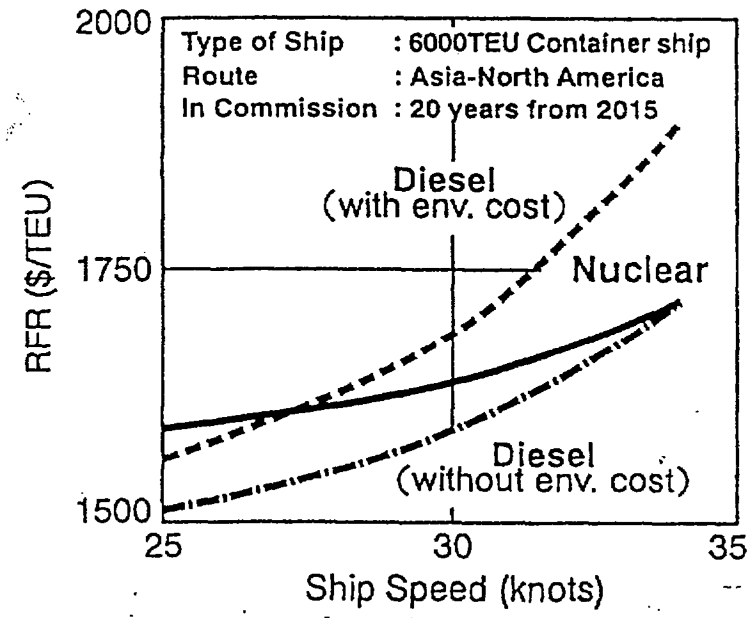


Fig. 10 Comparison of RFR with parameter of ship speed

The energy supply system can be designed to operate over a wide range of loads with various ratios of electrical and heat demands. There are possible cases as shown in Fig. 13. The case 1 is for electricity generation alone. The capacity of electricity generation is 100 MW, which will be sufficient for household use of 100,000 person. The case 2 is for electricity generation and heat supply. The energy for heating by hot water or steam will be more effective than electricity in a cold region or winter season. For this condition, the extracted steam from the outlet of the high-pressure

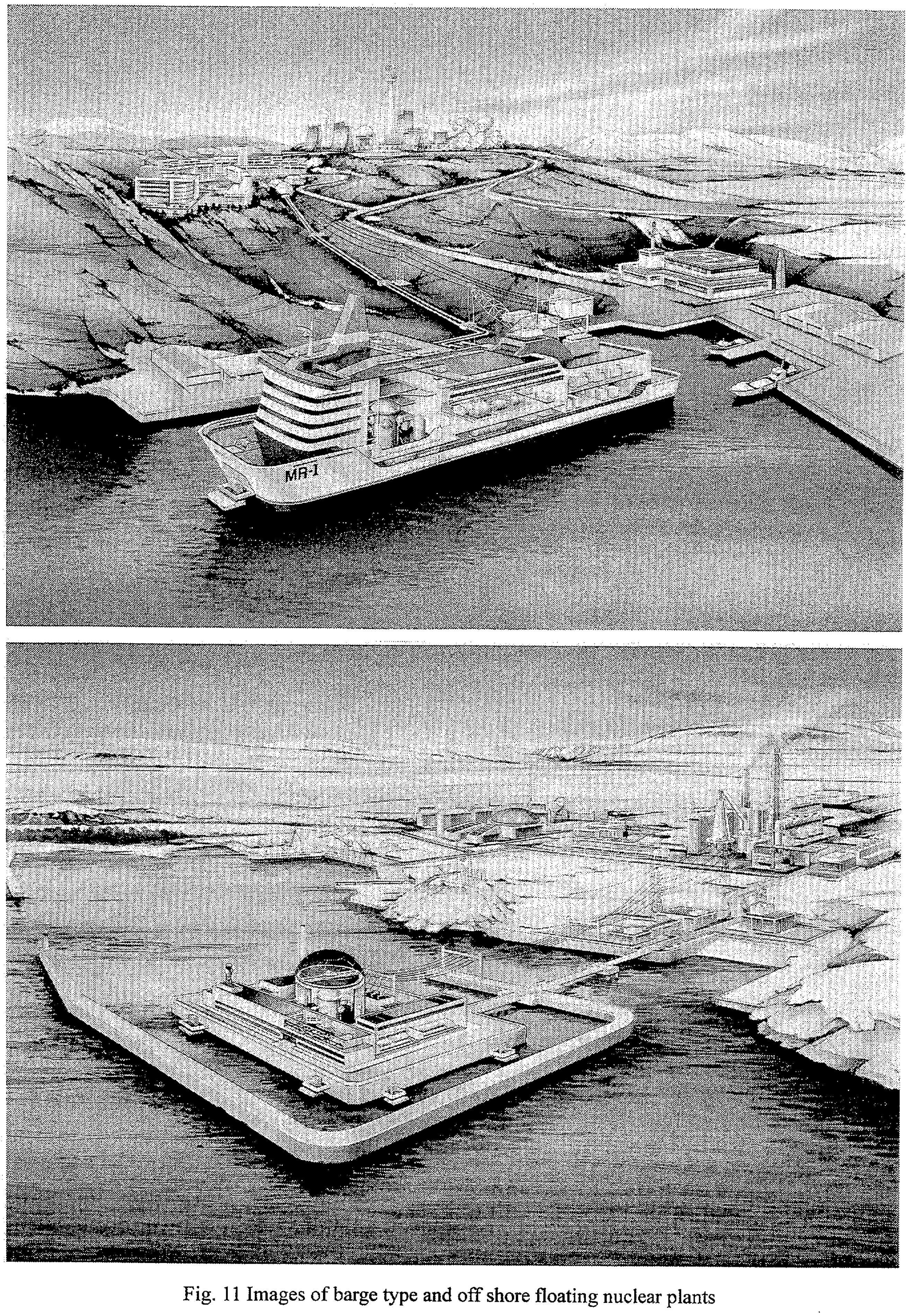


Fig. 11 Images of barge type and off shore floating nuclear plants

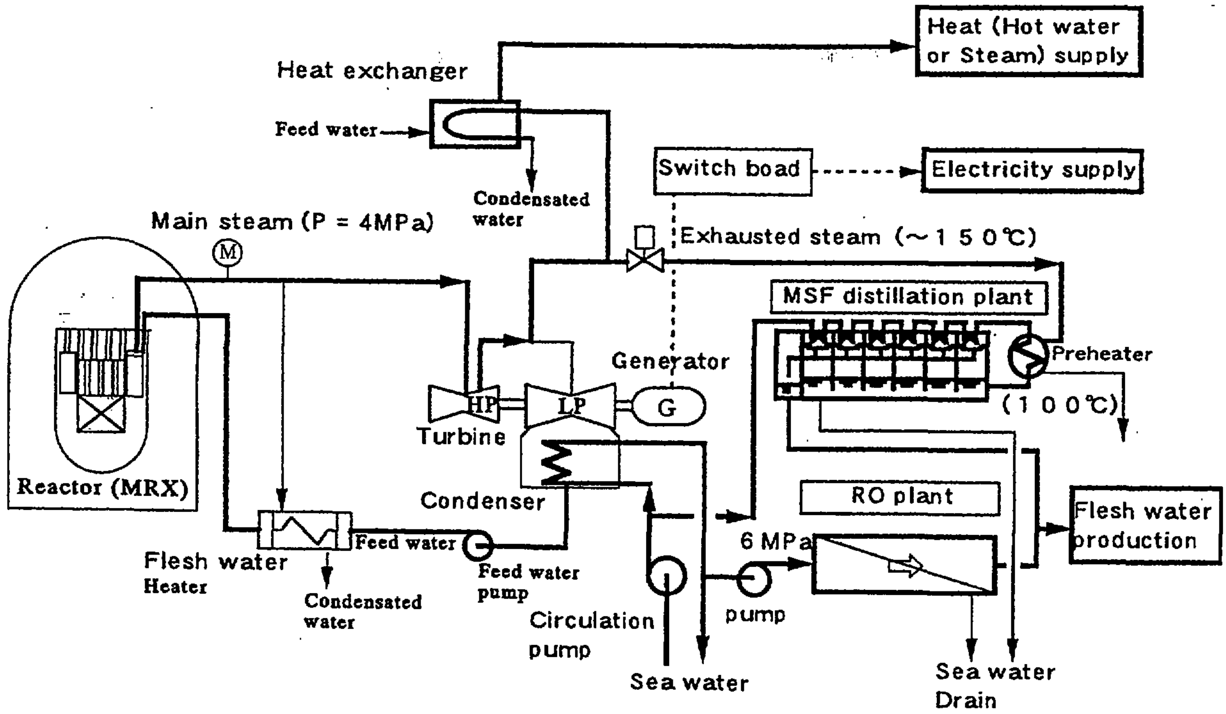


Fig. 12 Concept of nuclear energy supply system

(Case 1 : Electricity generation only)

Time =	0:00 – 9:00	9:00 – 17:00	17:00 – 24:00
Electricity	50 MWe	100 MWe	50 MWe
Reactor power	150 MWt	300 MWt	150 MWt

(Case 2 : Electricity and heat co-generation)

Time =	0:00 – 9:00	9:00 – 17:00	17:00 – 24:00
Electricity	50 MWe	83 MWe	50 MWe
Heat supply	50 MWt	50 MWt	50 MWt
Reactor power	200 MWt	300 MWt	200 MWt

(Case 3 : Electricity, heat and freash water co-generation)

Time =	0:00 – 9:00	9:00 – 17:00	17:00 – 24:00
Electricity	30 MWe	83 MWe	30 MWe
Heat supply	50 MWt	50 MWt	50 MWt
Fresh water (35,000m³/day)	30~50 MWt		30~50 MWt
Reactor power	180~200 MWt	300 MWt	180~200 MWt

Fig. 13 Variations of energy supply for electricity, heat and fresh water co-generation

turbine will increase at daytime and electricity generation will be reduced. The case 3 is for electricity generation, heat supply and seawater desalination. Since electricity will be consumed mainly in the daytime and less in the nighttime, fresh water production can be made in the nighttime. The fresh water can be produced by combination of the MSF and the RO about 35,000 m^3/day, which is sufficient for household use of 100,000 person on the base of 0.35m^3/day/person. The reactor power is required to vary in a day according to demand of energy utilization. The MRX can respond easily to power change.

Design features of the system are as follows.

- The reactor of MRX, light-weighted and compact integral type PWR is applicable.
- For protection of heat consumer, the three-circuit scheme is used with the pressure boundary between circuits.
- The MRX can be operated smoothly for load change according to demand of energy usage.
- Maintenance or refueling can be made in a short period with the one-piece removal of the reactor system as described in the previous chapter.

3.2 Nuclear Barge

A nuclear barge with the MRX can be sited in protected water area such as artificial bay of sea or river. The barge, special non self propelled ship can be transported by tug boats by water to the region of deployment. The concept of nuclear barge is shown in Fig. 14. The reactor thermal power is 300 MW to generate the electricity of 100 MW. The barge is designed to supply the electricity, but it can also supply the hot water or the fresh water by installing plants according to demands.

The displacement of the barge is 13,000 tons. Layout of the barge is designed as follows:

a) The reactor and electricity generation compartments are placed near the center of the barge to stabilize and trim,
b) The control room, the resident rooms, the radioactive waste process compartment etc., are laid at the bow, and the compartment for electric transformer and switch-gear at the stern,

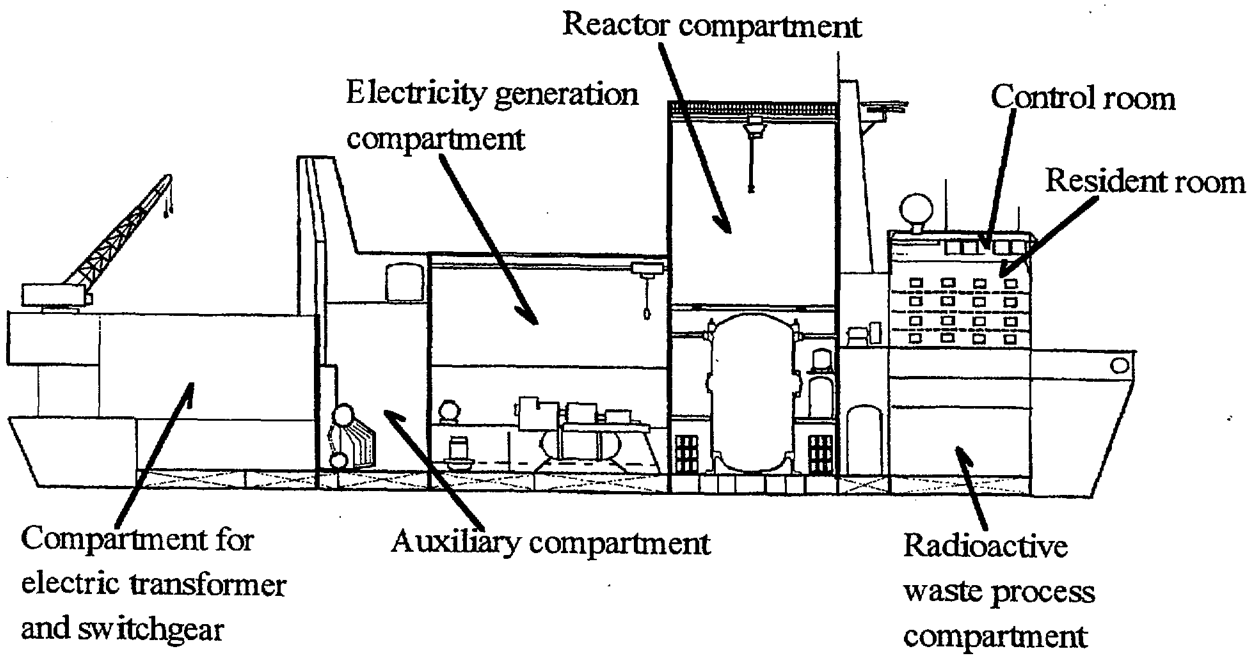


Fig. 14 Concept of nuclear barge with MRX

c) Upper part of reactor compartment is used for carrying of the fuel and an open space for overhaul of the reactor system,
d) The boiler and auxiliary electricity supply system is set at the auxiliary compartment for maintenance or the periodical inspection.

Mooring of the barge is one of important design subjects and depends on the condition of site. A possible mooring facility is made by a hydraulic shock-absorbing mechanism using the dolphin which can be used for large floating structures.

Maintenance and refueling of the reactor system can be done by the way of the one-piece removal of the reactor system at a dockyard, and replaced with another one.

4. CONCLUSION

(1) For a light-weighted and compact reactor, the radiation shield structure is critically important. The weight of MRX is reduced greatly by eliminating the secondary shield due to adoption of the water- filled containment. The integral type reactor is also effective for a compact reactor. Compactness of MRX is obtained by adoption of the integral type reactor with arrangement of almost all components inside the RPV.
(2) The engineered safety system of MRX is greatly simplified by adoption of the water-filled containment and the passive decay heat removable system relied on the natural circulation. Comparing the numbers of the sub-systems and equipment in the engineered safety system with other nuclear plants, the number of is decreased significantly.
(3) The MRX simplified passive safety system can ensure safety of the reactor, which is confirmed by the safety analysis. It is noteworthy that the MRX never uncover the core by help of the water-filled containment even in a LOCA.
(4) Simplification of the system can also contribute to improvement of reliability. Reliability of the MRX system is confirmed by the PSA and found the total core damage occurrence frequency is two orders smaller than that of existing PWR plants on land.
(5) The MRX can be used for not only a ship but also a wide range and multi purpose. The nuclear energy supply system with the MRX of 300MWt can provide enough electricity, fresh water to the population of 100,000 person.
(6) The nuclear barge with displacement of 13,000 ton is introduced as a possible application for energy supply.

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CIVILIAN APPLICATION OF PROPULSION REACTOR IN INDONESIA

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Abstract

It has been learned that to cope with energy requirement in the remote islands and less developed regions of Indonesia small or very small nuclear reactors producing electricity and/or process heat could be appropriately applied. The barge mounted propulsion power reactors are the attractive examples so far envisioned, and technology information of which is being exposed to the world these last years. The solutions for least maintenance and no on-site refueling, no radioactive discharge, and no radioactive waste to remain in the user country are among the attractions for further deliberations. It has been understood, however, that there are many uncertainties to overcome, especially for the developing countries to introduce this novel application. International acceptance is the most crucial, availability of first-of-the-kind engineering, prototype or reference plant that would prove licensibility in the vendor's country is the second, and economic competitiveness due to very small size is the third among issues to enlighten. The relevant regulations concerning marine nuclear safety, and of marine transportation, and proliferation of information, as well as international forums to justify the feasibility of related transfer of technology are the items that the IAEA could help to provide to smoothen any possible international transaction. Indonesia supports this AGM as one of the appropriate IAEA efforts on this line, and expects from it positive international consensus and possible studies/R&D work that this country could participate.

1. INTRODUCTION

In the national industrial plan, numbers of industrial centers are being developed throughout Indonesia. The industrial centers located on the Jawa Island are able to rely on the available infrastructure for on-land transportation means and electricity supply by the Jawa-Bali grid. Whereas those on the big islands as Kalimantan and Sumatra, the industries are able to rely on the primary energy sources (oil, gas, or coal), which are indigenous on their island. But those industrial centers, located more distantly from the energy sources will have to depend on the energy transported and delivered, which are expensive unreliable.

It has been learned that to cope with energy requirement in the remote islands and less developed regions of Indonesia small or very small nuclear reactors producing electricity and/or process heat could be appropriately applied. The barge mounted gas-turbine power plants, constructed in Surabaya shipyard, have been operating so far to provide temporary power for Balikpapan city. By size requirement, a nuclear propulsion reactor is the most attractive example so far envisioned. A modest quantity of technology information on floating small NPPs have been obtained and learnt.

PWRs and BWRs were earlier being developed for naval and civilian purposes in the industrially developed countries. Whereas in Russia naval pressurized water reactors have been known since longer time to be utilized in icebreakers and container ships. Earlier this year, presentations of Russian representatives were explaining these technologies for stationary (in-land) as well as floating power plants of PWR type to the BATAN and general audiences in Indonesia.

The first information on civilian design of liquid-metal fast reactor on barge was introduced to the authors of this paper in 1996 during the week of ICENES-96 Conference here in Obninsk.

This Pb-Bi cooled fast reactors, which previously used for submarines, are later to be designed on barge and said to be capable of providing electricity in full power for 4-6 years continuously. Advantages and improvement being pursued are mentioned in the references.

2. REQUIREMENTS

Short version of User Requirements on Small and Medium Reactors for Indonesia was presented in the IAEA conference in Dubrovnik, June this year [1]. It includes also requirements for small and very small size reactors, for production of electricity and possible use of process heat, sited land based or floating, for deployment in remote areas. It covers technical, safety, economic and financial requirements and general consideration typically applied for Indonesia, a developing country. Below are some points, which are related to the topics of this AGM.

2.1. Siting

Figure 1 shows map of Indonesia with centers of industrial activities. The less developed or remote areas are dipersed mostly at the eastern part of Indonesia. Industrial zones to be developed in the East Regions of Indonesia cover such areas as Bima (West Nusa Tenggara) as center of agro-business and agro-industry, Mbay (East Nusa Tenggara), Benaviq (Betano-Natarbora-Viqueque, East Timor). Whereas the industrial zones Pare-pare (Cental Sulawesi), Manado-Bitung (North Sulawesi) is to be developed as industrial centers based on marine resource, as well as agro-business and agro-industry.

Figure 2 shows the map of Indonesia with isoseismic lines and zones of subduction. Some isolated or less developed areas, which are at the East Regions of Indonesia may be located in the high seismic areas. The isoseismic lines shows for the Nusa Tenggara Timur, South of Maluku and Irian Jaya the horizontal ground acceleration are between 0.05 to 0.25 g, whereas for Sulawesi and North of Maluku are at the less end of the span.

Therefore, the barge mounted installation that would stand to the seismic disturbances and possible related tsunamis, as well as for simplicity in the remote area construction and operation would be preferable. Although in-land, stationary types, if they are assured complying the seismic requirements as the floating ones, are well appreciated.

As characteristics of the potential sites as well as their load and other technical demands may differ from site to site, detail requirements on design characteristics vary substantially. It is, therefore, significant that information on "envelope design characteristics" of proposed plant, i.e. the span in which the plants are designed, also be presented, so as to simplify the appropriateness of a site.

2.2. Possible options

Due to limited information on other designs at the authors' side, only two Russian reactor types are mentioned in this paper: the pressurized water KLT-40C and the liquid-metal (Pb-Bi) fast reactor Cruse-50 variants. As mentioned above the floating/barge version would look to be promising for remote areas or less developed regions [2-6].

The features of long core life (couples of years), minimal on-site construction, least maintenance and no on-site refueling, long campaign time, low environmental radioactive discharge during operation and no refueling at site are announced to contribute to simplicity of application.

By the build-own-operate (BOO) scheme the vendor would tow the NPP barge, with cool shutdown reactors in it to the construction site. And at the end of each operating campaign the vendor would replace the NPP with new one and tow back fully used cores

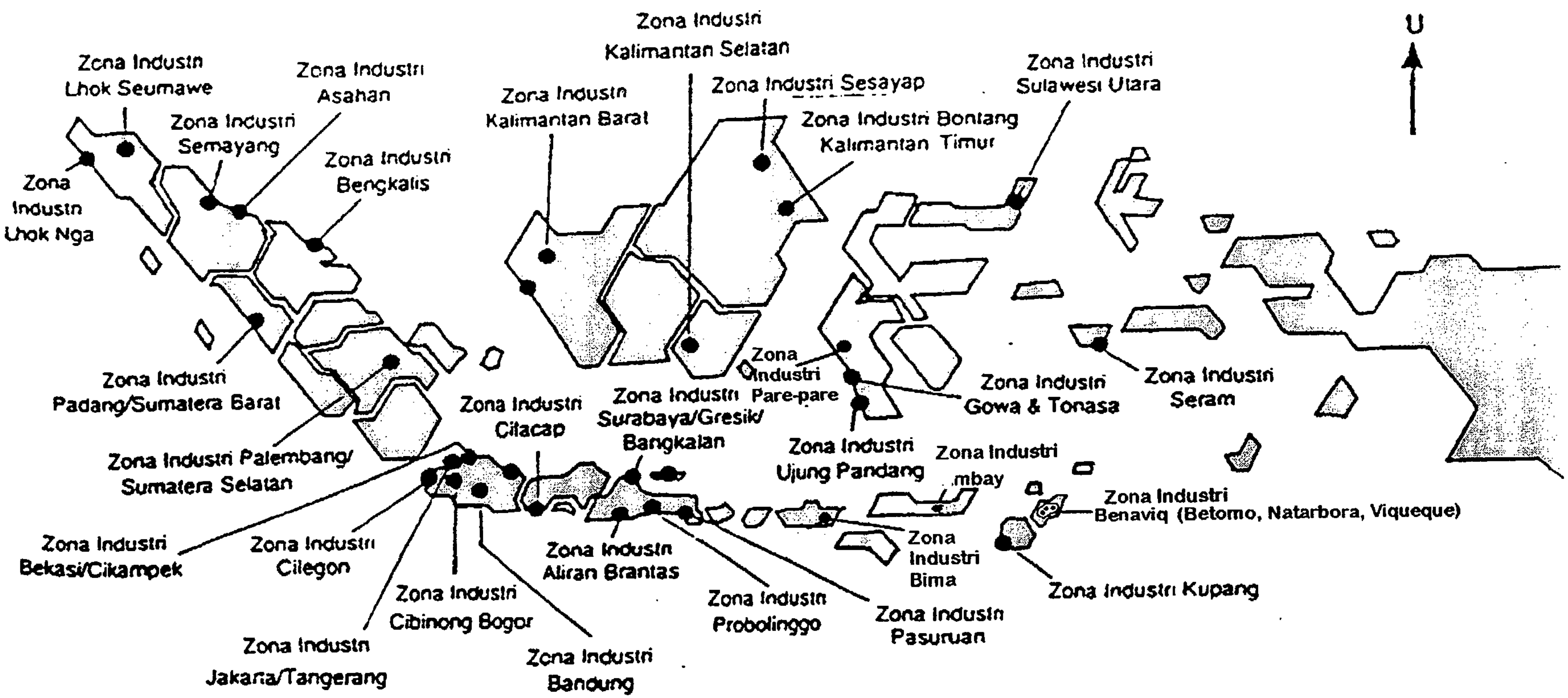
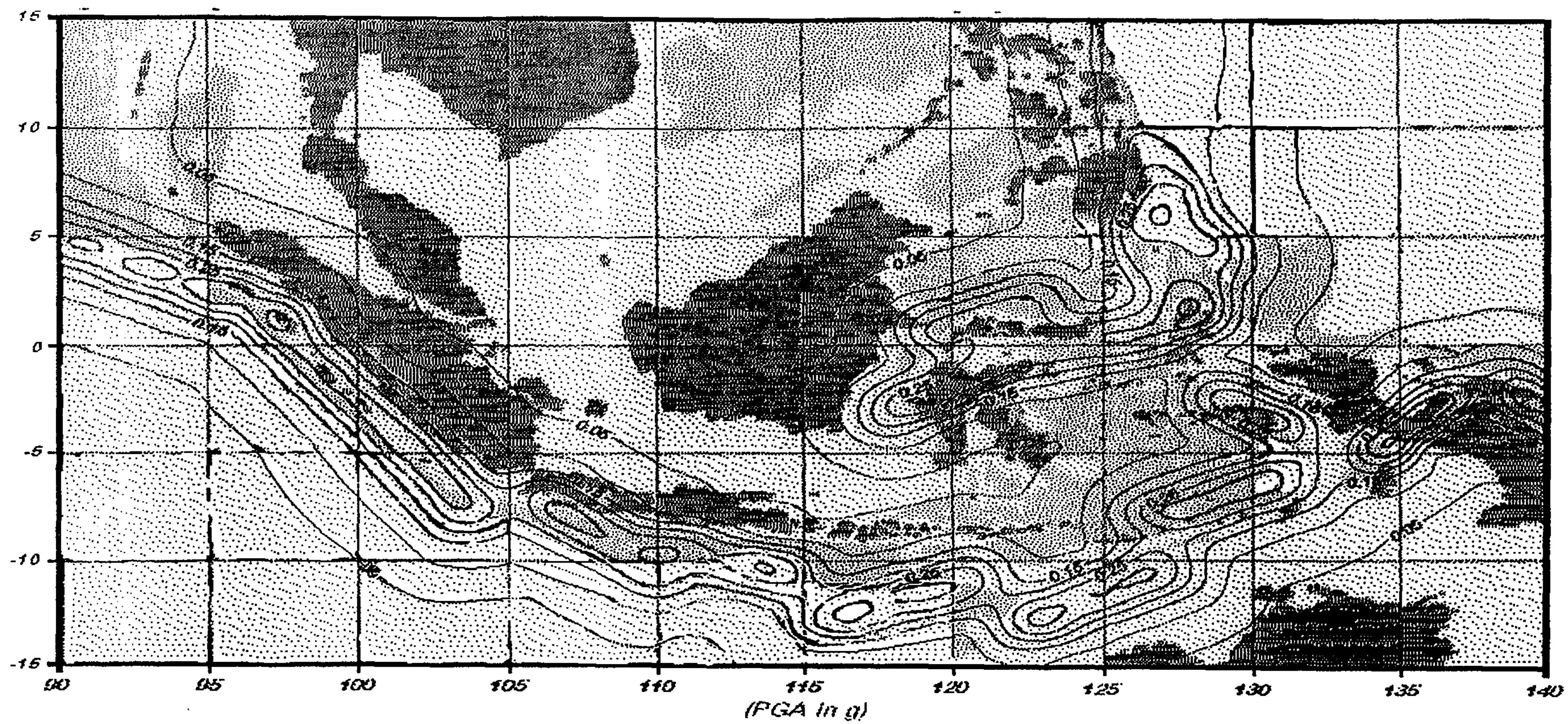


Figure 1: Map of Indonesia with industrial centers, existing and being developed [12].



Figrue 2: Map of Indonesia with isoseismic lines and subduction zones [13]

to the home country. In this way no irradiated fuels and radioactive wastes would remain in the user country. And this is one of the most attractions for further deliberations.

These advantages gained in 160 reactor years of KLT-40C operation [5] are derived from an advanced core design requiring uranium fuel with enrichment more than 20% U235 (HEU), and/or plutonium fuel. Some efforts, however, to reduce the enrichment to less than 20% (LEU) are being exercised, which would result an entirely new core.

The information on the KLT-40C reactors in its possible deployment in Pevek or near St. Petersburg has been widely publicized. On the other hand since the liquid metal fast reactor type has not proceeded into the market, it remains to the BATAN the least informed. It is anticipated that recent progress will be presented in the forum.

2.3. Licensibility in the country of origin

In the Indonesian Atomic Energy Act there is no specific difference in the present licensing steps for large-, medium-, small- and very small-sized nuclear power plants. The adoption of passive safety concept, non-active components, and other improvement of engineered safety features will be taken into consideration in making possible simplification of the licensing process.

The SMRs including the nuclear desalination plant (or process heat installation) shall be licensable in the country of origin. Meanwhile, to simplify the licensing process in Indonesia, adoption of passive safety concept, non-active components, and other improvement of engineered safety features should be taken into consideration as much as possible in designing the plant.

2.4. Economic criteria

Economic competitiveness due to very small size is the third among issues to enlighten. It is understood that small and very small power reactors has certain advantages in general (such as adaptation to small grid, less financing required, more manageable for domestic participation, etc.[7]), but the cost of generated electricity per KWh is usually high. While for the deployment at remote areas or at the less developed regions the economic criteria for the Small and Very Small Reactor alternatives in principle remain the least generating cost, but additionally the following are considered:

- the largest social gain,
- zero or the least government subsidy, and
- smaller than the cost to upgrade the infrastructure and transportation means in order to remove the "remoteness" qualification.

These competitiveness criteria are applied, for both conventional financing, and for BOO or other non-conventional financing schemes (e.g. with barter supplement). In principle an evaluation based on "profit and risk sharing" is considered.

2.5. Domestic participation

As usual requirement in the developing countries, maximal utilization of locally available materials, manufacturing capabilities and readily made products, as well as Indonesian nationals for labour, skilled workers and supervisory services are encouraged for the construction, operation and maintenance works, as far as they meet the specified quality. Indonesian companies are encouraged to participate and take the transfer of technology. But in the case where the plant would operate under a BOO scheme, in which the Independent Power Producer (IPP) alone would own the floating module of nuclear plant including nuclear fuels, the domestic participation is likely limited to the on-land facilities: switchyard, transmission, distribution and other supporting facilities.

These on-land facilities should be assigned with participation to the Indonesian companies. Whilst operation and maintenance participation would be optimized by the IPP itself. Local non-nuclear maintenance obviously could be executed by local shipyard or engineering companies. Other options of domestic participation could be expected in the design participation of Indonesian engineers in the vendor country, including for example the design of interfaces and non-nuclear auxilliaries.

2.6. R&D Cooperation

Participation in the R&D on certain design verification, first-of-the-kind engineering or the one having ultimate objective in the constructing of a prototype or demonstration plant in Indonesia might be of possible option. The user contribution in this case may cover the R&D site, available manpower, licensing and accompanying public relations. Later experiences would benefit all participating sides and create regional market of their own. With this scheme of cooperation a process of gaining licensibility or a reference plant might be aimed.

3. INTERNATIONAL ISSUES

3.1. Acceptance

It is understood that the highly enriched fuels and/or plutonium were used to achieve compact configuration of the core as well as long core life without refueling at the plant site. If the highly enriched uranium fuels be used, therefore, international utilization of this special materials should be put forward into an international consensus, before any developing country could step forward. On the other hand, when the LEU conversion from the HEU have been readily obtained the newly designed core would need proveness in the operating experience.

International acceptance is the crucial issue that one country should foresee, since it involves non-proliferation and safeguards. In possible scheme of Build-Own-Operate (BOO) the whole barge with the nuclear fuel will stay and remain within the supplier's possession before, during and after the use in the reactor. This might solve the problem of ownership of such materials within the territory of a user country, unless otherwise does not accord to the local regulations.

The extent of security that required for the safeguarding highly enriched uranium and/or plutonium fuels within the plant and during transport through international waterway would likely need international consent. Similar consideration would apply to the movement of floating reactors having whole irradiated cores containing full spectrum of radioactivities.

3.2. Assistance

The relevant regulations concerning marine nuclear safety, and of ocean nuclear transportation, are the least exposed to the Indonesian entities. Other topics relate to any particularities, e.g. of the liability and insurance coverage of such floating plants, compared to stationary ones recently discussed in Dubrovnik conference [10]. The facts that on the BOO scheme the owner of the reactor & fuel, operator on one hand and the utility on the other hand may belong to different country with respective own regulations.

There have been long lists of IAEA information highlighting and promoting the various ways of SMR's application; some of them quoted here as reference [7-11]. Moreover there have been numbers of IAEA expertise being sent to developing countries to assist in evaluating possible options of nuclear power plant introduction. Nevertheless, further assistance of expertise and dissemination of technical information that once was non-civilian, as well as international forums to help assess the feasibility of related technologies are the items the IAEA could organize.

4. CONCLUDING REMARKS

The application of civilian version of nuclear propulsion reactors in Indonesia looks realistic for the coming intermediate terms although some uncertain points shall be clarified. Indonesia supports this forum as one of the appropriate IAEA efforts on promoting propulsion nuclear reactors for civilian application, and expects from it positive international consensus on various points discussed above. Possible studies or R&D works that this country could participate in are anticipated.

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NEW CONCEPTS FOR SMALL POWER REACTORS WITHOUT ON-SITE REFUELLING FOR NON-PROLIFERATION

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Abstract

The report discusses the preliminary requirements and a technical approach for developing small reactors and the rationale for selecting them. It then discusses the four nuclear system technologies and how they might proceed to meet the requirements. Brief discussions are provided on the approaches to stimulating the appropriate international and industrial participation necessary to finance development of a design with improved proliferation resistance that is useful to the developing countries.

EXECUTIVE SUMMARY

Energy demand in developing countries is increasing to support growing populations and economies. This demand is expected to continue growing at a rapid pace well into the next century. Because current power sources, including fossil, renewable, and nuclear, cannot meet energy demands, many developing countries are interested in building a new generation of small reactor systems to help meet their needs.

The U.S. recognizes the need for energy in the developing countries. In its 1998 Comprehensive Energy Strategy, the Department of Energy calls for "research into low-cost, proliferation-resistant, nuclear reactor technologies" to ensure that this demand can be met in a manner consistent with U.S. non-proliferation goals and policies. This research has two primary thrusts: first, the development of a small proliferation-resistant nuclear system (i.e., a technology focus); second, the continuation of open communication with the international community through early engagement and cooperation on small reactor development.

A system that meets developing country requirements must:

- achieve reliably safe operation with a minimum of maintenance and supporting infrastructure
- offer economic competitiveness with alternative energy sources available to the candidate sites
- demonstrate significant improvements in proliferation resistance relative to existing reactor systems

These challenges are the most significant driving forces behind the LLNL proposed program for development of a new, small nuclear reactor system.

This report describes a technical approach for developing small nuclear power systems for use in developing countries. The approach being proposed will establish a preliminary set of requirements that, if met, will cause new innovative approaches to system design to be used. The proposed approach will borrow from experience gained over the past forty years with four types of nuclear reactor technologies (LWR, LMR, HTGR, and MSR) to develop four or more pre-conceptual designs. The pre-conceptual designs will be used to confirm the adequacy of the requirements through iteration and trade studies. A down selection to a preferred and backup concept would be made following a 12-18–month design effort. The selected designs, system design specifications, and the necessary R&D programs would be developed in greater detail over the next three and one-half years.

resistance, and will maximize the reliance on passive safety features to reduce the risk of serious accidents and their consequences, simplify operations and maintenance, and reduce the need for the developing country to establish a sophisticated and expensive nuclear infrastructure.

In particular, to eliminate all on-site refueling, the reactor will be equipped with a long-life core that will be returned to the supplier when spent. This process will be managed under international control to further both overall non-proliferation objectives and to reduce the infrastructure burden on the developing country. It will also reduce the anticipated burden and expense to the International Atomic Energy Agency for assuring security associated with expanded international use of nuclear energy. An integral part of the program will be the development of new approaches for implementing international safeguards applicable to the entire fuel cycle including recycling and waste disposal.

1. INTRODUCTON

Energy demand in developing countries, particularly in Asia, is increasing rapidly to support growing populations and economies. These trends are projected to continue well into the next century. This demand will be met by a combination of fossil, renewable, and nuclear sources. However, the current mix of these energy sources does not, and cannot, satisfy all demands for new energy. Fossil fueled sources add to the accumulation of greenhouse gases and are of limited supply in many areas. Renewables, including hydro, are in limited supply, or are at an insufficient level of technical readiness to play a major role in meeting energy demands. Today's nuclear systems are too large and expensive for many areas, especially those lacking the necessary institutional infrastructure, expertise, capital or large established power grid. A small nuclear power system could provide a credible solution for meeting the growing energy demands of many regions.

U.S. foreign policy on nuclear issues is dominated by proliferation and nuclear safety concerns. Recent events in India and Pakistan have served to refocus world attention on the threat of proliferation. We are proposing to develop a new nuclear system that meets stringent requirements for a high degree of safety and proliferation resistance, and also deals directly with the related nuclear waste and spent fuel management issues.

Development of a small, modular nuclear power system offers promise as a viable alternative to either the capital intensive large nuclear systems or the environmentally problematic fossil fuel systems. Our vision is to develop a system featuring unique design elements that overcome many of the barriers to implementation of nuclear power systems in many parts of the world. Elimination of on-site refueling through a combination of long-life cores and small systems with replaceable modules reduces proliferation concerns, especially those related to spend fuel storage. Small size, transportability, modularity and the requirement for a high degree of autonomously safe operation reduce infrastructure and construction requirements of the host country, and contribute significant reductions in initial capital and operating costs. Small unit power levels make the system appropriate in areas lacking the large electrical grids normally associated with today's large nuclear systems.

A recent survey completed by the International Atomic Energy Agency (IAEA)[1] not only confirmed the need for such systems, but projected that by the year 2015, developing countries will need 70–80 small and medium-sized reactor systems. Currently, several countries are developing small reactors for this potential commercial market. For example, South Korea has been developing a concept known as SMART; Argentina has been developing a concept known as CAREM. Both systems are based on down-sizing today's large light-water reactor technology to make it suitable for small power levels. China is developing a small, high-temperature gas-cooled reactor for seawater desalination and process heat. Russia is proposing several small reactor systems based on marine propulsion experience for use in developing countries, as well as remote regions of Russia. The efforts of these countries tend to focus on the development of the reactor without integrated considerations of the overall fuel cycle, proliferation, or waste issues.

2. DESCRIPTION OF SCENARIO SELECTION PROCEDURES

The system being proposed is targeted for developing regions such as South East Asia, Sub-Saharan Africa and South and Central America. There are countries within these regions with high population growth with goals to raise the standard of living. The electric power grids in some of these areas are minimal. In addition, because of geographical isolation or lack of indigenous alternatives, nuclear power may be a viable alternative. Several countries (Russia, South Korea, Argentina, and China) are developing concepts for small systems that could be used in these areas.

Projected population growth for these countries will be well over 250 million by 2025. For their standard of living to improve to half the average of the developed countries, they will need approximately 1kw of new generating capacity per person. This means that they will need approximately 250 Gwe new capacity, the equivalent of 250 1000-MW plants, or 2,500 100-MW plants. Continued growth in population and standard of living will double this estimate by 2050. Much of this capacity will be met with conventional technologies, including large nuclear systems. Even with large systems in place, the IAEA estimates that 70–80 small and medium-sized reactor systems will be required by 2015. With even greater growth anticipated beyond 2015, the potential outlook for such systems is optimistic, and could require production rates on the order of 10 or more plants per year.

Successful development of a small reactor system requires a comprehensive systems approach that considers all aspects of manufacturing, transportation, operation, and ultimate disposal. This approach has been used previously in the development of a U.S. space nuclear power system, with consideration given to many diverse requirements such as no planned maintenance, highly reliable autonomous operation for a long period of time, no refueling, and ultimate disposition. While the engineering designs used for space nuclear power systems are not directly applicable to terrestrial applications, the systems approach itself offers insight into the development of a new terrestrial system suitable for developing countries. The U.S. aircraft industry also uses a systems approach in the production of new aircraft that may be suitable for small reactor modules.

Applying this system design concept to development of small nuclear power systems requires that planners consider the entire system lifecycle, including waste issues. This will lead to significant reductions in the generation of radioactive waste for the system as a whole, where, with other methods of planning, one waste minimization activity often increases wastes farther along in the system.

By considering the complete fuel cycle and impeding access to the fuel by eliminating on-site refueling, there appears to be a potential for significantly improving proliferation resistance. Such a system would simplify safeguarding the fuel and could provide a basis for development of new policies for addressing the storage and disposal issues of the fuel cycle.

3. SYSTEM REQUIREMENTS

Our principle objective is to develop a system that provides an affordable, safe and secure energy supply to countries with critical energy needs but insufficient resources to support more traditional energy solutions. The following requirements outline how this objective may be achieved. The "system concept" can be specified by key system requirements, including:

3.1 Reactor delivered pre-assembled and pre-fueled

The nuclear reactor should be delivered to the site pre-assembled and pre-fueled. If shipping a pre-fueled reactor is precluded by safety or technical considerations, fueling must be accomplished under strict international control. Procedures to seal fuel in the vessel and eliminate further on-site access into the vessel must be established.

3.2 Highly autonomous operation

All operations, from initial startup to final shutdown, must be as autonomous as possible. Operator functions could be limited to pressing a button to start the nuclear system and performing those duties necessary to operate the power conversion or other non-nuclear systems. For the most part, an operator should only be required to monitor the status of the system. Advanced instrumentation and control technologies, coupled with the anticipated improvements in inherent reactor control mechanisms in these low-power systems, makes potential realization of such goals possible. It is recognized that this requirement is very demanding and it may become more complex as the program progresses. Significant cost savings will be associated with reductions in the number of highly trained staff required at the site.

3.3 Reliance on passive safety mechanisms

All credible failures should be safely terminated (1) by passive mechanisms in the nuclear system and (2) without the release of radioactivity to the power system site. Postulated severe accidents should be terminated without requiring emergency off-site responses. Plans for site recovery following such postulated situations should also be identified. Passive safety mechanisms are a necessary corollary to achieving the anticipated staff reductions.

3.4 Limited planned maintenance

Planned maintenance will be limited to non-nuclear, or electrical and control components easily accessed outside the reactor enclosure. Special attention must be given to eliminating instrumentation that is inherently short lived because of temperature or radiation damage.

3.5 Replacement and disposal of the reactor and its contained fuel must be integrated into system design

No on-site refueling means that at the end of a reactor core's life, the entire reactor module is replaced. This feature will help ensure that nonproliferation goals are met. A careful design will incorporate the replacement, reconditioning and disposal of expended reactor modules, including disposition of spent fuel.

These demanding goals are likely to be achievable only with systems of relatively low power compared to the 1000MWe plants commonly used today, with the added benefit of greater security, safety, and public acceptance of the expanded use of nuclear power within developing countries.

We recognize that one of the major challenges will be to accomplish these goals with an economically viable system. Today's economic approach to nuclear power is through economies of scale. Our concept approaches the economic issue from a different perspective: we rely on the economics of mass production, coupled with cost savings achieved from dramatically reduced on-site installation, operation and decommissioning costs, reduced site infrastructure requirements, and simplified "type" licensing to overcome the loss of economies of scale.

While many of these goals have, in principle, been met by space power systems, practical terrestrial systems that might meet this challenge will be orders of magnitudes larger than space systems, but may need to be one to two orders of magnitude smaller than the current LWRs. Even so, these requirements—while challenging—are not impossible; other complex systems such as a modern gas turbine or combined cycle plants meet many of them, including standardized factory fabrication, simple installation and startup, and highly automated operation.

4. SAFETY CONCEPT

Two very demanding safety requirements have been identified. First, all credible accidents (those considered to be within the design basis) should be safely terminated by passive mechanisms in the nuclear system. By "safely terminated" we mean not only that the consequences of such events are limited to the site, but such events must not result in life-limiting damage to the systems or radiological consequences to the on-site staff. Second, postulated severe accidents (those beyond the design basis) must be terminated without requiring emergency off-site responses. These postulated events should be consistent with known physical principles, and include operational failures coupled with postulated failures of protective features and equipment. An approach to recover from such postulated events, one that permits ultimate recovery of the site, should also be identified. That is, bounding accidents should not result in loss of the use of the site due to contamination. These two requirements are key to achieving the operational and design simplifications desired in the small nuclear system.

The small unit size and low unit cost of this approach presents a unique opportunity to perform a much broader range of safety verification testing than is afforded by conventional nuclear designs. This level of testing would provide a full-scale demonstration of the ability of the design to achieve the required safety envelope. Certification of the design by testing, including severe accident testing, would contribute to broad acceptance of the system in developing countries. Such an approach also supports licensing certification by test as once proposed to the NRC[2].

Safety certification of the design would be completed by the supplier, similar to procedures with commercial aircraft. Licensing the use of a factory-fabricated standard design in many different countries would then be similar to approving the use of certified aircraft. The regulatory burden on the user country would be substantially reduced.

Reliance on a high degree of autonomous operation, achieved through a combination of passive, inherent physical responses and advanced control systems, minimizes the potential for human error initiating or exacerbating off-normal events or accidents. It also reduces the training and regulatory burden on the user country. Far fewer nuclear specialists are necessary to approve and operate such systems. Similarly, eliminating the need for on-site refueling and minimizing on-site maintenance further reduces the need for a supporting infrastructure in the host country; such maintenance services may be handled by a single supplier service organization consisting of very highly trained and skilled professionals who would provide essentially all repair and maintenance functions for the entire fleet of these standardized facilities.

5. OPERATIONAL CONCEPT

The infrastructure necessary to support conventional nuclear power development is very expensive, and beyond the resources of most developing countries. One of the primary goals of the proposed approach is to reduce the need for such an infrastructure. The system requirements of highly autonomous operation, simplified and minimized system maintenance, and elimination of all on-site refueling all significantly contribute to this goal.

To some extent, the complex nature of today's large-scale nuclear energy systems is a result of the many support and safety systems, often redundant, needed to ensure the safety of these very large, high-power systems. Many of these systems can be reduced or eliminated with smaller systems, where natural forces (such as natural circulation flow) are sufficiently strong to satisfy many safety and operational criteria. A more aggressive approach to automated operations is being proposed to support both large reductions of on-site staff and the technological infrastructure necessary to support operation, maintenance, and regulation. If the very high levels of reliability and simplicity of operations can be achieved, it may be possible to reduce the nuclear staff to a couple of people per shift in the main control center and a minimal electrical maintenance staff.

Nuclear instrumentation and control systems present a greater challenge. Even though it may be possible to develop a system that is inherently controlled by temperature for all normal operations, it appears that it will always be necessary to include control devices and instrumentation in the nuclear system to adequately control and monitor startup from cold shutdown, total shutdown, and possibly to compensate for fuel burnup. Design configurations that reduce the number and complexity of these instrumentation and control systems and components need to be developed.

Typically, the conventional power conversion components of a plant include dynamic machinery and numerous active control mechanisms. It will be nearly impossible to avoid planned periodic maintenance on this equipment. Even though it might be desirable to simplify and reduce the staff and infrastructure needed for this portion of the plant, the technical expertise and infrastructure required for effective maintenance and operation of these systems could be shared with that needed by other, more conventional power-generating systems the developing country will also have. Depending on the specific configuration of the system, however, there may be a cost or operational benefit that may demand supplier servicing of selected components within the power conversion portion of the plant.

6. MANUFACTURING

The importance of a totally new approach to manufacturing of the small reactors, not just modularization, cannot be over-emphasized. The concept of "designed-in manufacturability" can result in significant cost savings, particularly when amortized over a number of systems. The goal of producing a system that can be easily transported and installed can also contribute to manufacturability, as this goal leads directly to components having more "reasonable" size and mass than most of today's large nuclear power systems.

An aggressive approach to factory assembly of the plant and shipment of modules that can be quickly installed at the site will be necessary to keep costs within an acceptable range. The factory production line will need to be designed as an integral part of the product—similar to the approach taken by Boeing Airplane Company to produce their 777 model.

The plant design would be based on production of many hundreds of essentially identical modules. Ideally, three or four major assemblies would be fabricated, quality inspected, and tested prior to shipment. With exception of the nuclear assembly and its auxiliary systems and components, these types of major production assemblies have been demonstrated in the large diesel and combined cycle plants operating under much the same technological complexity as the proposed nuclear plant.

Simplification of the system design coupled with factory manufacturing of standard modules may make it possible to challenge the economy of scale with the economy of mass production. As discussed later, it will be necessary to identify a large stable market to support this approach.

7. DELIVERY AND INSTALLATION CONCEPTS

Transportation and installation of barge- or ship-mounted small nuclear power systems is clearly feasible. Designs for barge-mounted "conventional" nuclear systems was near reality in the early 1970s, and even today, ship-mounted small reactor concepts are being proposed.

However, it is also desirable develop a system concept that is also amenable to land-based siting, including the need to transport the modules over land. This desire will place additional constraints on the size and mass of the individual modules making up a full system, especially when one considers that supporting transportation infrastructures in the developing countries (roads, railways, etc.) may be substandard or even non-existent. These considerations strongly suggest that design of special transport and handling equipment and procedures be integrated into the overall system design.

While it is likely that barge- or ship-mounted systems can be optimized at larger powers than their land-based cousins, we believe the advantages provided by mass production of very standardized units will eliminate any minor cost advantage a larger system might enjoy for barge-mounted applications.

The transport systems for removing the spent reactor modules present the greatest challenge. The problem of shipping a highly radioactive component either on land or by sea must be addressed, and associated environmental concerns will influence how the nuclear system is designed. For example, capability for recovery of a sunk shipment from the ocean bottom may be a requirement, and could have a strong influence on the nuclear module designs, even for those installed in barges or submersibles. The possible need for additional shielding, especially for overland transportation, could have an impact on the size of the nuclear module originally installed and could cause it to be quite small.

Another important requirement, related to the removal transportation, is the time period permitted between final shutdown and required removal. Because of the intensity of radioactivity and decay heat in the nuclear module immediately following shutdown, it would be preferable to leave the shutdown module in place for many months. While potentially unavoidable, such a postshutdown delay would likely complicate replacement of the reactor module, particularly for sites with limited real estate. Although there may be a slight reduction in proliferation risk by removing the spent nuclear module soon after shutdown (as opposed to later), we believe that the difficulty associated with removing or accessing a spent reactor module to be a sufficient deterrent. Rapid removal/replacement of a spent reactor module might be accomplished sooner with the barge- or ship-based systems, but unless the installation allows separation and replacement of the nuclear system, it would mean a significant investment in unused power conversion equipment during transport or refurbishment. This may not be a great penalty if the on-station time for the system is tens of years.

These and many other broad system issues will be the topics of the trade studies necessary to identify the preferred system. It is clear that manufacturing, delivery, and installation will have an important impact on size selection of the reactor system; the size may vary depending on the system technology.

8. RECYCLING AND WASTE DISPOSAL

One of the important aspects of our systems approach is the integration of waste disposition and component recycling into the design of the overall system, including materials selection and mechanical design. It is our intent that the nuclear reactor module or the entire plant (a possibility in the case of barge-mounted systems) be shipped to an internationally monitored site for refurbishment of useable components, recycling of highly valued materials, and the preparation of waste materials for permanent disposal.

The characteristics of the "refurbishing" site and its facilities must also be integrated into the overall design, and will take into account spent fuel processing, waste treatment, and disposition of other materials and components. The extent to which other equipment is refurbished may also influence the character of the site. A barge-mounted system might be entirely refurbished; therefore, the site design would need to consider this entire refurbishment activity.

One of the more difficult issues associated with recycling and waste disposal is not a technical issue; rather, it is a political and institutional issue of how to deal with wastes, especially nuclear waste, resulting from operations in a foreign country. The U.S., and most other countries, have restrictions on the acceptance of nuclear waste from other countries. It is possible that new national and international policies, treaties, and laws may be necessary to gain all the advantages that a centralized approach might afford.

9. SYSTEM SAFEGUARDS

Nonproliferation safeguards will be part of the proposed system to provide (1) assurance that facilities and materials are not used for illicit purposes, and (2) timely warning of an event taking, or having taken place. The proliferation risk associated with a nuclear system derives from the following four issues:

- attractiveness of fissile materials,
- accessibility of the materials,
- utility of the facility for illicit purposes (for example, irradiation of fertile materials), and
- size and character of the infrastructure needed to support the nuclear system.

Traditional large-scale nuclear power systems are quite proliferation-resistant. The fresh LEU fuel is of too low enrichment to be directly used in a weapon. The reactors are ill-suited for illicit irradiation and production of weapons material. Plutonium in spent fuel has poor isotopics for weapons applications, and is inherently protected by the significant radiation field arising from the fission product inventory. Even so, safeguards of LWR plants are needed because none of these barriers to proliferation risk is, by itself, completely effective. Diverted fresh fuel could be used to reduce the enrichment effort, given appropriate facilities. Fertile materials could, with difficulty, be irradiated in LWRs. The radiation barrier inherent to spent fuel decays with time, and plutonium from LWR spent fuel is considered a weapons-useable material, even if not ideal.

The elimination of on-site refueling directly attacks the two greatest proliferation risks of the traditional power reactor: accessibility of materials and use of the facility for illicit purposes. Elimination of on-site refueling removes easy access to both fresh and spent fuel from the reactor site. Fissile material is found only inside the reactor, where it is protected by both limitations of physical access and a very intense inherent radiation barrier. The only period where fissile materials might be considered at risk is during transportation and set up, and during early operation where the fission product buildup is limited. Access to fissile materials and use of the reactor for illicit irradiation is further complicated by the lack of physical features and infrastructure to open the reactor vessel.

In addition to the direct reduction of proliferation risk resulting from the elimination of on-site refueling, the need and complexity of on-site surveillance can also be reduced. With the absence of on-site fresh or spent fuel storage, the need for active and/or periodic monitoring of fuel stores is eliminated. The very lack of a refueling infrastructure implies that the time necessary for opening the reactor vessel to gain access to fissile materials would be so long that the very act of shutting the reactor down may serve as the primary alert indicator. This offers potential for simplified safeguards and security that might only require monitoring to ensure the reactor is indeed operating. This could even be accomplished or augmented remotely, perhaps using satellite platforms to monitor the IR signature of the facility.

The high degree of automation desired for these systems offers three additional safeguards. First, such autonomous operation offers opportunities for remote monitoring and diagnostics that may be exploitable to assist safeguards monitoring. Second, such design offers the potential for reductions in system transients and resulting shut-downs. In such an environment, any off-normal event becomes a potential safeguards alarm. Finally, the combination of autonomous operation and elimination of on-site refueling reduces the infrastructure, including personnel, available to support illicit activities.

10. TECHNOLOGY OPTIONS

This section summarizes a preliminary assessment of the design and technological approaches that might be taken to achieve the requirements. The discussion is based on a preliminary assessment of the strengths and weaknesses each technology offers in meeting the requirements, and a cursory look at how these strengths and weaknesses might be exploited or mitigated. This assessment is only intended to provide an approximate prioritization of the concepts; each concept will be reviewed in greater detail as part of the overall program. The technologies are addressed in their current order of preference, based on their likelihood of meeting the no on-site refueling requirement with practical limits on development cost and schedules.

10.1 Liquid metal cooled systems

Liquid metal cooled reactors (LMR) have been under development for more than 40 years, and several have been built and operated on commercial power grids. Sodium has been generally used as the coolant in these systems, and experience with this technology has been mixed. Sodium leaks have been the most notable technical issue, and the fact that sodium is very chemically reactive with air and water has contributed to most of the concerns about LMRs. The technology is clearly still developmental even though there is considerable valuable experience.

Concepts based on fast LMRs have high internal conversion of fertile to fissile material and therefore have the potential for long core life without initially high reactivity that requires some form of poison compensation. LLNL and the University of California have developed some variations on the Japanese 4S concept that could extend its neutronic life to more than 15 years [3, 4, 5, 6]. Although there are unanswered concerns about the clad and structural damage from large neutron irradiation, the results of preliminary work are encouraging.

Fuel systems for LMRs are well developed. Oxide, carbide, nitride or metal fuel have been demonstrated and provide many options for addressing the long life requirement for the fuel. Metal fuels, in particular, offer promise of improved safety characteristics.

Liquid, metal-cooled systems operate at low pressure and have very large thermal margins relative to their boiling temperature. There is negligible thermal energy stored in the coolant available for release in the event of a leak or accident. This translates to the potential for compact system designs, including containment. In addition, the large thermal expansion coefficients of liquid metal coolants inherently support good thermal characteristics for natural circulation cooling and provide very favorable passive safety characteristics.

As mentioned above, the sodium coolant used by most LMR developers has created some safety concerns because of its chemical reactivity with air and water. Lead, lead-bismuth or other alloys of lead appear to eliminate these concerns because the chemical reactivity of this coolant with air and water is very low. The Russians have extensive experience with this coolant and, when combined with the Japanese 4S configuration, it conceptually is a very attractive system. Several major issues must be addressed, however, including what appears to be the most important: materials corrosion. This is a serious issue, and although the Russians have had success in this area, long-term materials compatibility must yet be established.

Extensive development and demonstration work, beyond that required for a small LWR-based concept, may be required. These reactors also require fuel that is enriched to nearly 20% and therefore fuel recycling may be necessary to achieve economic performance. Because of the developmental uncertainties and the possible need to include fuel recycling in the system, the cost of implementing a system design based on liquid metal coolant is uncertain.

10.2 Gas-cooled Systems

Gas-cooled reactors, although not extensively used for electric power generation, have continued to be used and receive developmental attention. There have been many years of successful power operation on the power grid in England. Helium has become the coolant of choice in the high-temperature versions of this technology. Recently, development work has focused on modular reactors coupled to gas turbines to improve both economics and the passive safety characteristics. Considerable effort has been invested in development of graphite-coated fuel particles that are imbedded in a graphite matrix. However, there is also considerable experience with cermet fuels which have fuel particles imbedded in a metal matrix. This later technology has not received much commercial attention, although extensive development is completed for application in aircraft and space systems.

There is some continuing interest in gas-cooled reactors in South Africa, China, and Russia. ESKOM, the South African state-operated utility, is interested in a high-temperature, gas-cooled reactor combined with a direct cycle gas turbine for powering rural areas that are currently without electricity. They have developed a preliminary design for a system based on the pebble bed reactors developed in Germany. General Atomic and Russia, in cooperation with others, have also completed a study on design of a gas-cooled gas turbine plant for use in Russia. China has operated a 5 MW_t gas-cooled reactor and has plans for construction of a 200 MW_t which is intended for process heat applications.

Since gas-cooled reactors are predominantly thermal spectrum reactors, the challenge will be to develop a concept for extending core life. De-rating the fuel and operating the reactor at lower power densities than under optimum operating conditions is one possibility. This could have an unfavorable impact on economics. Thorium fuel cycles applied to these types of reactors could be used to contribute to increased core lifetime. However, on-line refueling—in which pebbles are added and removed on a frequent basis—has been the emphasis in the pebble bed reactors. Some conceptual designs for pebble bed reactors specify that the reactor vessel is only partially filled at the start of life. Fuel is gradually added, thus extending reactor life. Consideration could be given to hardening the spectrum and improving the internal conversion, possibly even using cermet fuel in a fast spectrum reactor.

The modular designs studied in recent years have very favorable passive safety characteristics. Both the Russian and ESKOM designs incorporate robust passive safety capability. Although the systems run at high pressure, there is little energy stored in the single phase coolant and the thermal margins in the fuel are very large. Thus, a loss of coolant during an accident may be accommodated passively. However, containment design continues to be a question and may add to system size and complexity.

The direct cycle gas turbine is the most attractive configuration for the gas technology. Inclusion of steam generators contributes complexity and decreases efficiency and reliability. Even though there has been no operational experience with a direct cycle gas turbine HTGR (and therefore development costs and risks are high), once developed it is expected to be very economical. This is a viable candidate that has seen considerable progress in the development phase but awaits demonstration.

10.3 Light Water Cooled Systems

The LWR is the most highly developed and deployed reactor and fuel cycle system in the world. Many small versions have been designed and even deployed, although most have been limited to submarine versions and other special purposes where economics has been a secondary requirement. The possibility of adapting various highly developed marine systems to small electric generator designs is being seriously considered as a viable option.

The main challenge facing this technology will be to extend the refueling interval much beyond two years while maintaining the inherent safety objectives. Power de-rating, similar to the gas reactor, can extend core life at the expense of increases in power cost and physical size. There are efforts underway to extend fuel life to three years, but to make this technology viable, it is likely that fuel life will need to be extended further. Long-term autonomous control may also be a challenge, partly due to reactivity changes over the life of the plant and other technical issues, such as the need to maintain water chemistry and pressure control.

The small units can be made simpler and include large margins to safety. Natural circulation cooling at full power is possible. However, because of the large amount of stored energy in the coolant, a pressure-retaining containment is necessary to meet the safety objectives. It is also likely that active safety components such as isolation valves may be required.

LWR technology is the most developed technology and therefore may have very low development costs. Yet the significant development needed to achieve long fuel lifetimes required here may offset the economics afforded by the mature underlying technology base. Additionally, the high system pressure and significant thermal energy stored in the coolant complicates safety and containment design, and is expected to increase system size, mass, and cost.

10.4 Molten Salt Systems

One molten-salt reactor system has been constructed, and although its experimental operations were successful, there have been no others. The interest has been largely academic, but conceptual designs of commercial plants have been developed. This technology would require the largest development and demonstration program, and with no licensing experience, could prove to be a difficult option to implement.

One of the major concerns about this technology, besides the limited technology base, is the lack of materials with demonstrated long-term compatibility. Long-life operation will require on-line chemistry control and processing of the molten-salt fuel, and although demonstrated at the laboratory level, such processing remains a development issue.

Interest in the molten salt system is based on the promise of long reactor life through automated fuel processing and management. Automated additions of fuel and removal of the fission products, if possible, could eliminate fuel life as a limiting factor to system life. Such automation is clearly a major challenge.

Because the reactor can be designed with very little excess reactivity in the core and the molten salt has very good heat transport characteristics, the potential for achieving passive safety objectives also exists. The fact that (1) the molten-salt reactor is a low pressure system, and (2) the coolant is very chemically stable and does not react with air or water also support the passive safety characteristics of this concept.

However, the fact that the fuel is both mobile and unclad raises serious concerns about fuel leakage and containment and control of the gaseous fission products. These concerns will impact the designs of containment and fuel chemistry management systems, and may require significant developmental efforts to demonstrate.

This technology is clearly very speculative and would likely require the most development and demonstration. The cost and schedule for implementing a concept based on this technology may also be the most demanding. With innovative use of automation, however, it may be possible to realize a design concept that best addresses functional requirements.

10.5 Other Technologies

We have briefly reviewed other technologies, such as heavy water reactors, but have not identified any with sufficient merit to warrant consideration. The CANDU-HWR is well developed but uses on-line refueling which is contrary to the objective of excluding on-site refueling.

There are LWR and LMR advanced fuel-cycle schemes being investigated with potential to improve the proliferation resistance of current reactors. However, they usually require extensive fuel shuffling or fuel modifications to make the fuel less suitable for weapons application. These approaches may be found suitable for large reactors, but they do not appear compatible with small reactors that have a requirement for no on-site refueling.

11. R&D ISSUES

The proposed program for designing a new, proliferation-resistant reactor is based on systematic, requirements-driven evaluations. That is, we believe it imperative to develop a consistent, well thought set of performance, design, and operational criteria that truly reflect the goals of the program rather than to make marginal improvements to existing systems. Even though the latter approach has clear short-term advantages in terms of development effort, cost, and schedule, we believe the marginal improvements in both proliferation resistance and fundamental safety are insufficient to support broad institutional acceptance of such designs.

While it is possible that systems based on existing technologies may be conceived that meet the objectives, it is likely that an extensive R&D effort will be required. Without clearly defined concepts, we can only outline the general scope of such a program. In fact, the development of system concepts is the first major program objective.

Clearly, systems designed for long-term operation without refueling will place new demands on fuels and in-core materials compatibility and performance, and will demand improved analytic techniques, including nuclear analysis and structural performance methods. Desirability of transportable designs may require new methods of manufacturing, transport, and installation. New approaches to siting and design certification, perhaps through new international institutions, could facilitate a broader acceptance of such systems and installations, and minimize infrastructure development in the host country. Implementation of highly reliable automated controls that rely on a minimum of instrument signals will need to be developed.

12. APPROACHES TO FINANCING

Financing the initial research necessary to realize a small reactor concept for developing countries is under discussion with the U.S. Department of Energy (DOE). We are seeking to recruit international advisors early in the program, and build international participation as the program matures and funding increases. The program being proposed would include two major phases lasting approximately 12 years. The first five-year phase would lead to a conceptual design of a preferred and backup system. The first phase would be financed by the participating governments. During the second phase, a prototype would be developed for certification testing. During this phase, it is expected that support for certification testing would come from major governments, while support for establishing the production facilities would come from commercial participants. These facilities would be used to produce the prototype as well as the subsequent production models. It is envisioned that the program would shift from largely government-supported laboratory research into a partnership with industry that would then lead to industry-operated facilities subject to international monitoring. These facilities would include factories and production lines used to produce standard assemblies suitable for use at many sites.

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USE OF RUSSIAN TECHNOLOGY OF SHIP REACTORS WITH LEAD-BISMUTH COOLANT IN NUCLEAR POWER

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Abstract

The experience of using lead-bismuth coolant in Russian nuclear submarine reactors has been presented. The fundamental statements of the concept of using the reactors cooled by lead-bismuth alloy in nuclear power have been substantiated. The key results of developments for using lead-bismuth coolant in nuclear power have been presented.

1. INTRODUCTION

In the beginning of the 1950s nearly at the same time the USA and USSR launched the development of the nuclear power installations (NPI) for nuclear submarines (NS). In both countries the work was carried out for two types of NPIs: with pressurized water reactors and reactors cooled by liquid metal coolant (LMC).

In the USA sodium was chosen as LMC as it possessed the better thermo-hydraulic characteristics. The ground-based test facility-prototype of the NPI and the experimental nuclear submarine "Sea Wolf" were constructed. Yet the operation experience has pointed that the choice of coolant which was chemically active with respect to oxygen and water has not justified itself. After several sodium\water interaction, the RI was decommissioned together with the compartment and replaced by the pressurized water RI.

In the USSR the lead-bismuth eutectic alloy was chosen as LMC [1].

In the USA the scientific and research development works were conducted on using lead-bismuth coolant (LBC), but the alternative to solving the problem of corrosion resistance of structure materials and maintenance of coolant quality (the coolant technology) did not give any positive results and those works were stopped.

In our country the comprehensive problem on coolant technology, structure materials corrosion and mass transfer has been solved as a result of systematic work of several organizations (IPPE, TSNII KM "Prometey", EDO "Gidropress", OKBM, NITI and some others) for about 15 years. (Some failures happened in early days of mastering the new technology). The solution to this problem has ensured long and reliable operation of NPIs using LMC at the NSs.

Proceeding from the experience gained from design and operation of NPIs using LBC, there have been developed both the concept of exploiting the reactors cooled by LBC for nuclear power plants (NPP) and a number of proposals for using LBC in nuclear power (NP).

2. THE ANALYSIS OF EXPERIENCE OF OPERATING THE REACTOR INSTALLATIONS USING LEAD-BISMUTH COOLANT AND THE ACCIDENTS HAPPENED

When the last NS with the RI cooled by LBC was removed from the Navy, the specific phase of ship nuclear power development had been completed. The innovative nuclear power technology which could not be comparable with any other one in the world was demonstrated to our industry.

Due to ongoing developments of the RIs using LBC for civilian NPPs the experience gained needs for the thorough analysis and control in these designs in order to make the best use of the LBC advantages (high boiling point, chemical inertness, etc.) and to minimize the effect of its disadvantages (melting point is ~ 125°C).

The operation experience also accompanied by the number of accidents which are inevitable for any new technology mastering (the history of technique has demonstrated it) and revealed the difficulties of servicing these RIs at their base places and refueling has received inconclusive assessment by the experts, who have been familiarized with it more or less.

Some authors, who have been familiarized with this experience only by "hearsay", allege in their memoirs that for the NSs adoption of Projects 705 & 705K (class "Alpha" according to the NATO terminology) of RIs using LBC was a tragic error.

Below there is the presentation of the key results of operating the RIs using LBC, the results of analysis of the deadly serious accidents happened, causes of the difficulties in servicing the RIs at the base places and fuel reloading and at the other phases of RIs life time, measures which have been realized in the designs of the RIs using LBC and which eliminates the accidents causes and operation difficulties. This presentation enables us to make an unbiased assessment of the experience gained [2].

2.1. The Analysis of the Accidents Happened and Difficulties of RIs Base Serving

In the course of design and operation of RIs using LBC there were accidents at three NSs, that was the cause of impossibility of further NSs operation. These were the RI accident at the left side of the NS of Project 645 in 1968 [3] when the core was melted partially, the OK-550 RI accident at the NS of Project 705 (task order 900) in 1971 when the additional pipelines of the primary circuit lost their tightness and the BM-40/A RI accident at the NS of Project 705K (task order 105) in 1982 when the global corrosion damage of steam generator (SG) pipe system of the water-steam circuit happened and there was about 150 l spill of radioactive coolant into the compartment [2].

One of the difficulties of RI servicing at the NS base places and refueling is the necessity of continuous steam ingress into the steam heating system (SHS) of the primary circuit in order to provide the liquid form of coolant and join up periodically the RI with the base installation to perform the maintenance works on coolant technology.

As far as the RIs of the NPPs are concerned, the problem on coolant liquid form maintenance is not so urgent because of existing the outside power sources and stationary arrangement of RIs.

Below there is given the description of the accidents mentioned in the course of their progress, the analysis of their origin causes and technical measures on their elimination.

2.1.1. The RI Accident at the NS of Project 645.

It was the only accident the cause of which is concerned with LBC using. As a result of low studying physical and chemical coolant operation processes, the substantiated specifications on impurity compositions in coolant, instrumentation for coolant quality control and equipment to provide the maintenance of required coolant qualities in the course of operation have been lacked. This level of LBC knowledge can be compared with that of water coolant when it was allowed to convey water from the water pipeline into the steam boiler.

As a result of that operation, uncontrollable accumulation of significant masses of lead oxides in the primary circuit happened, they could have formed when the pipelines of the primary circuit gas system, which were necessary for its repair, were depressurized, and thus the air penetrated into the primary circuit. Besides that, the primary circuit was contaminated by products of oil pyrolysis, which was the working medium for seals of rotational shafts of pumps that provided the gas leak proofness of the primary circuit. Masses of oil were spilled into the primary circuit because the oil seals had not been reliable enough.

When the rate of SG leakage increased suddenly (it had started some time before the accident), the oxides accumulated and other impurities filled the core, that was the cause of the violent decline of heat removal. Negative temperature reactivity effect was the cause of transfer the automatic power control rod up to the upper switch terminal and spontaneous power reducing to 7% of nominal one. This was the first symptom of the accident.

But the operational documentation did not include any necessary instructions for the operator how to act when that kind of situation arose. Instead of resetting the emergency protection (EP) at the left side reactor, he followed the commander's directions (it occurred in the course of navy training) and tried to maintain the given power level by continuous removal of compensative rods (CR) out of the core. All reactivity reserve for 12 CR was released in about 30 minutes, though it was intended to provide for the power reserve generation about 4000 efficient hours. When CR were removed, the fuel in the local core area, where heat removal was deteriorated, melted and left the core together with the coolant flow. Signals of radiation hazard in the compartment that called for shutting down the RI and removing the crew into the compartments being distantly removed from the RI were not taken into account.

After this accident the works on the coolant technology problem have been launched. For many years these works have been carried out at the number of organizations under the scientific supervision of SSC RF IPPE. As a result, the problem has been solved successfully and the many-year experience of the following RIs operation has corroborated it.

Later the NS of Project 645 was withdrawn from the Navy and after special preservation of the RI and reactor compartment sunk in the Kara Sea.

It is necessary to point out the main technical measures for eliminating the causes of such accidents:

- in order to eliminate the accumulation of oxides the maintenance of some excess inert gas pressure in the gas system of the primary circuit has been provided when repair works of equipment and fuel reload are to be performed. In order to eliminate the possibility of air penetration into the primary circuit and the radioactivity yield into the environment the most possible tightness of it has been provided. For this purpose special repairing and refueling equipment have been developed;
- the sensors of thermodynamic oxygen activity which enable to control the content of oxygen dissolved in LBC and detect alloy oxidizing at the very early phases have been designed and introduced;
- rejecting the use of the oil seals of the pumps shafts and adoption of water seals or gas-tight electric drivers of the primary circuit pumps. This eliminates the oil penetration into the primary circuit and contamination of LBC by the products of oil pyrolysis ;
- using the ejection system of high-temperature hydrogen regeneration that has been built into the RI and ensures chemical recovery of lead oxides by hydrogen (the explosion proof compound of helium and hydrogen is used) and enables, if necessary, to purify even hard contaminated circuit from lead-oxides;
- using the continuously operated system of coolant purification from irreducible impurities on the glass fabric filters;
- using the automatic system of coolant quality control which is equipped by sensors of continuous control of coolant quality and protective gas, ensures the preservation of oxide films on the

surfaces of the primary circuit structure materials contacting with coolant and eliminates their corrosion deterioration as well as ensures the early diagnostics of sabnormal states.

2.1.2. The RI Failure at the NS of Project 705 (Task Order 900)

Since the beginning of the RI testings in 1970 and its further operation in 1971 and in 1972, the RI operation has been accompanied by the higher content of moisture in the air of the tight compartment (TC) where the RI was mounted. The tests have demonstrated that the causes of moisture were bad air-tightness of the seal of one SG cover because of the flaw in the nickel gasket, that was changed then, and steam leakage through the steam heating system welds which have been made unsoundly, and there was no possibility to eliminate this leakage because of compact assembly.

As a result of cold surface sweating inside TC, forming water drops were the cause of wetting the heat insulator and "dry" protection materials which involved chlorides.The drops of water saturated with chlorides touched the primary circuit hot ancillary pipelines made from austenite steel and gave rise to their corrosion cracking on the outside surface that has been fully verified by the results of the RI inspection performed. Through corrosion damages of the primary circuit ancillary pipelines at two of three heatexchanger loops and impossibility of their repair because of compact assembly caused the decision of removing this NS out of the Navy and carrying out the RI inspection.

Thus, this accident is not concerned with LBC use. Similar accidents followed by the RI failure happened with pressurized water reactors when sea water touched the rustproof pipelines of the primary circuit.

Technical measures eliminating the causes of such accidents are the following:

In the designed perspective RI it has been used the pool type integral system of the primary circuit arrangement, which eliminates fully any primary circuit pipelines out of the monoblock unit vessel including comparatively thin-wall ancillary pipelines of small diameter, there are no valves. Therefore, ramified system of steam heating is eliminated. Fabricating the RI monoblock unit under the plant conditions ensures high quality and delivery of the reactor block available for operation. The integral arrangement almost completely eliminates the possibility of coolant leakage. Besides that, the use of safety vessel is provided. The looser RI arrangement at NPP ensures the better conditions for performing the assembly works and controlling their quality.

2.1.3. The RI Accident at the NS of Project 705K (Task Order 105)

Global corrosion damage of the SG evaporation sections pipes made from perlite steel occurred as a result of not meeting the requirements for the water-chemical regime (WCR) for feeding water of the SG. It was the result of the fact that under real operation conditions the way of reducing the oxygen content in feeding water by electron-ion-exchanging filter with copper-containing charge, which was provided by the Project, caused to copper escape into the secondary circuit that was the cause of severe electric-chemical corrosion of the piping system of the SG evaporation sections.

As a result of through pipes damage, steam from the secondary circuit began to penetrate into the primary circuit, where after separation from coolant it condensed in the emergency condenser (EC) specially provided in gas system in case of leakage in the SG. As internal volume of the EC had been filled by the condensate step by step, according to the signal the operator many times drained the EC by removing the condensate accumulated into the suitable reservoir, and thus he eliminated the essential pressure increase in the primary circuit gas system.

However, the EC drainage was stopped because of not clear reasons. Heatexchanging surface of the EC was completely flooded by water and condensation of steam penetrating stopped. The pressure increase began at the primary circuit gas system. The strength of the gas system and primary circuit could bear the full working pressure of the secondary circuit. That is why in that case there could not be any tightness loss of the primary circuit.

Nevertheless, the tightness loss occurred, and it was caused by the following. The gas pocket in the leakage reinjection pump (LRP) located in the pump tank below the LBC level had adjusting manometer with ultimate pressure of 4 kg per cm^2 . According to the instruction, if the RI was in operation, this manometer had to be shut off by the valve. The instruction requirement was violated and the valve occurred to be open. Due to this fact, when steam pressure in the gas pocket of the LRP tank reached ~ 6 kg/cm^2 and the LBC level in the internal pocket of the LRP was increased with corresponding increase of gas pressure, the sensitive manometer element destroyed, gas escaped from the pump pocket, and under the steam pressure that was the cause of filling up the gas pocket of the LRP by lead-bismuth alloy and its further leakage through the damaged manometer into the inhabited section of the reactor compartment. (The scheme is presented in Fig.1).

Radioactive air contamination by polonium-210 aerosols reached 10 MPC. Due to the following proper actions, the crew irradiation and radioactive contamination were within the permissible limits. The analyses of crew bio-samples, which had been performed by medical service, demonstrated that none of the crew had the content of polonium-210 more than 10 percent of the maximal permissible value.

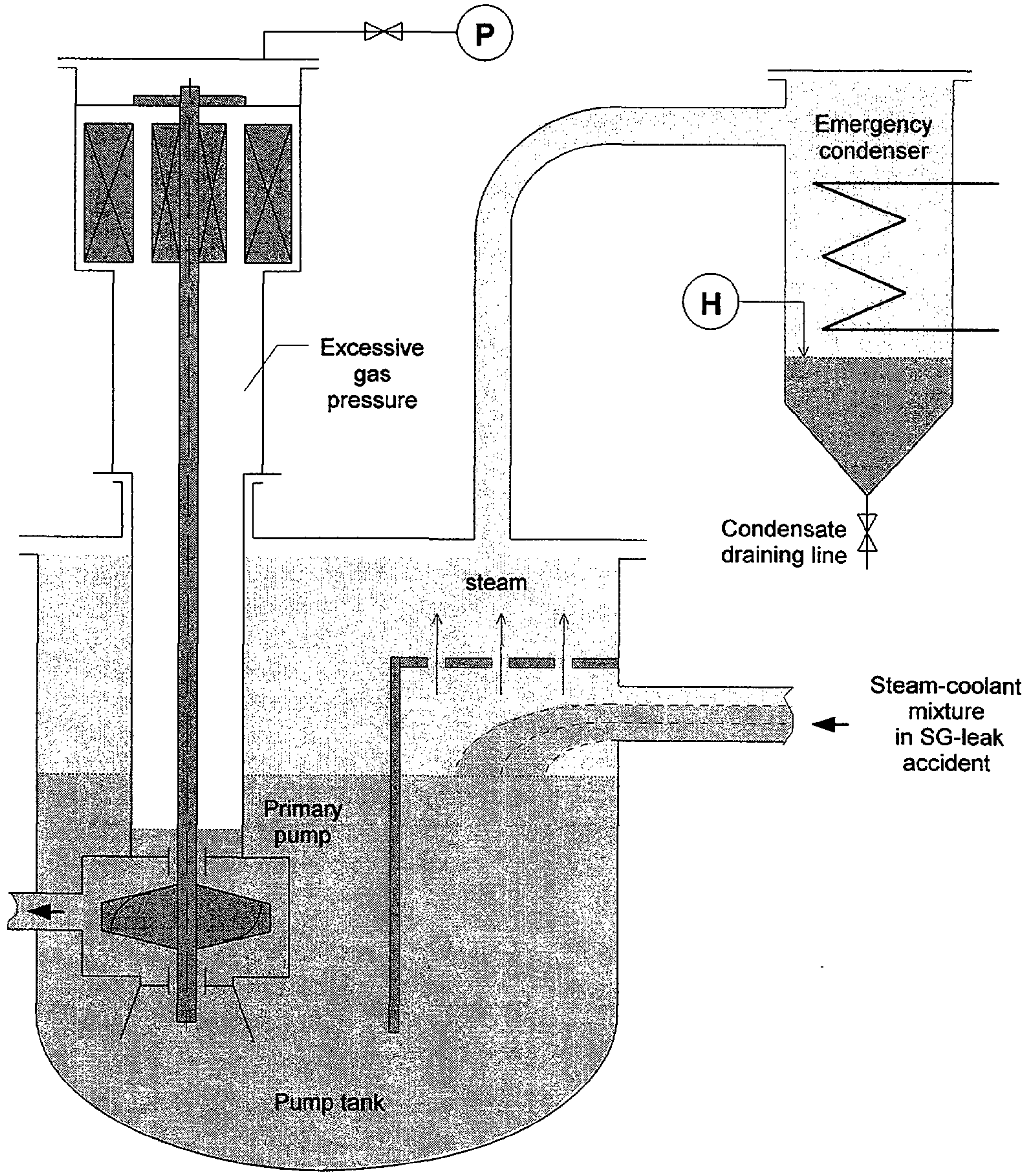


Fig. 1.

RI examining showed that it could be reconditioned. However, another decision was accepted. It was decided to change the whole reactor compartment of this NS by the new one fabricated earlier. The motive for this decision was the following. In the course of this RI fabrication at the Machine-building plant in Podolsk there was faulty change of the SHS pipes fabricated from high-nickel corrosion-resistant steel by the pipes made from common stainless steel of the same size.

This error was found out after the RI unit was fabricated and it was impossible to change the pipes. Because the service life of stainless steel pipes was restricted by corrosion conditions, there was accepted the decision to limit the service life time of reactor unit up to 25000 hours and fabricate the reserve RI unit in order to use it for changing off-spec one in the course of the NS overhaul period. In 1982 the service life of the SHS stainless steel pipes had to be expired, and that was the motive for changing the NS reactor compartment.

The analysis performed has demonstrated that the cause of this accident and the accident at the NS of task order 900 is not concerned with the use of LBC.

The following technical measures ensure elimination of such accidents at the new generation RIs:

- eliminating copper-containing materials out of water-steam circuit;
- using more corrosion-resistant steel as material for SG pipes under water -steam conditions instead of low-alloyed steel of perlite class;
- providing passive drainage of the EC when it is filling up by the condensate up to the given level;
- uniting the gas volume inside the pump electric motor with the total one above the free coolant level;
- providing the great extent of control-fitness and repair-fitness of the RI.

2.2. The Problems of Mastering the RIs Cooled by LBC

Among the key problems which have been solved in the course of design and operation of this type installations we must emphasize **the LBC technology problem** - i.e., development of systems and devices ensuring measurement and maintenance of the LBC quality required during its long-time operation both under normal conditions of leak-proof circuit and in the case of partial tightness loss of tightness of the circuit in the courses of repair works and reactor reload. Functioning those systems and devices is necessary for eliminating structure materials corrosion and slagging the circuit by lead oxides [4]. It should be pointed out that in the early days of mastering LBC, when the necessity of developing and implementing the measures on the coolant technology had not been realized, there were cases of reducing the coolant cross sections up to the full blockage of coolant flow rate because of depositing lead oxides and other impurities and all the resulted consequences (see 2.1.1.).

Corrosion resistance of structure materials have been ensured by using special steel alloyage, applying protective films to them in advance and maintaining necessary concentration of corrosion inhibitor - dissolved oxygen - in LBC [5]. The importance of these measures has been corroborated by the fact that when the necessary coolant quality was maintained, for several thousands of hours under the 650°C temperature there were no corrosion of fuel element steel cladding. However, when the dissolved oxygen concentration was inadequate (under the specially provided conditions), it took about 20 hours for through corrosion damage of the 5 mm thickness pipe under the same temperature.

Melting point of LBC is about 125°C. LBC maintaining in liquid state under all RI operation regimes is ensured by using the SG with multiple circulation over the secondary circuit, besides that, the inlet temperature of the water supplied to the SG is higher than LBC melting point. For initial heating-up and maintaining the primary circuit under hot condition at a low level of power release in the core the system of steam or electrical heating may be used.

The substantiation of the possibility of multiple coolant "freezing-defreezing" in the RI was an important practical problem. Low shrinkage of LBC during solidifying and rather high plasticity with low strength in solid state facilitate the elimination of RI damage when alloy is transiting from liquid to solid form and its further cooling down up to the ambient temperature. A special order of the temperature-time heating regime has been developed for safe RI "defreezing". This problem is dealt with in paper [6].

The specific feature of LBC is the formation of α-active polonium-210 radionuclide with a half-life of ~ 138 days when bismuth is irradiated with neutrons.

The major reason for its radiation danger is the formation of radioactive polonium aerosols when hot LBC contacts with air. It could happen under conditions of emergency tightness loss of the primary circuit and coolant spillage. In this case, as the RI operation experience at the NS has displayed, the yield of Po aerosols and air radioactivity (according to the thermodynamics laws) reduce quickly with temperature decreasing and spilled alloy solidifying. Fast solidification of spilled LBC restricts the area of radioactive contamination and simplifies its removal in the form of solid radioactive wastes.

Low polonium concentration in the coolant (at the level of 10^{-8} of at.%) and formation of thermodynamically proof chemical polonium-lead compound, which additionally reduces the polonium pressure by 1000 times, individual and collective protection personnel's facilities developed, methods for equipment decontamination and recording activity on the surfaces, methods of performing repair works have lead to that fact that for a long-term period of operating the reactor using LBC including the repair works of the primary circuit equipment and removal of coolant leakage (also in the case of LBC penetrating into the secondary circuit as a result of NS crew errors), there have been no cases of personnel extrairradiation by this radionuclide above the permissible limits. This positive practical result agrees with conclusions of foreign experts who have investigated the polonium hazard problem if LBC is used for nuclear reactor cooling [7], [8]. Paper [9] is focused on the analysis of this important problem

The following characteristics have been obtained in the course of NPI tests and operation: power and parameters of installation, the campaign lifetime, the reactivity margin, reactivity coefficients, poisoning effects, temperature distributions, dynamic parameters, coolant radioactivity, dose rates of neutron and γ-radiation behind the shield. They were in sufficiently good agreement with calculation results.

Among the positive properties of the RI using LBC, which have been discovered in the course of operation, one can point out the simplicity of control, high manoeuvrability and short time of reaching the power regime out of subcritical reactor state, the possibility of RI operation if there is small leakage in the SG pipe system, high repair-fitness of the SG by plugging the depressurized pipes, the possibility of RI stable operation at any low power levels, the possibility of quick changing the circulation regime of coolant with essential change of its flow rate, almost complete generation of designed power by cores under normal and acceptable conditions of leak-proofness of fuel rod claddings.

While performing repair works and reactor reloading, it is not necessary to carry out primary circuit decontamination which is concerned with collecting, storing, transporting and reprocessing masses of liquid radioactive wastes (LRW).

The cores and absorbing rods of the control and protection system (CPS), which have ensured total design power service, have demonstrated high operation serviceability.

During the last 10 years of NPIs operation there have been no problems with either structure materials corrosion in the primary circuit or the deviations from specifications on the circuit purity.

Experience of development and operation of NPIs using LBC at the NSs and ground-based facilities-prototypes enables to make a number of practically important conclusions on the arrangement and equipment of the primary circuit for reactors cooled by LBC for NPPs [10] .

The best engineering and economical parameters should be expected if the arrangement of the primary circuit equipment is integral one.

The most convenient design scheme of the SG is that one in which liquid metal circulation is performed in the interpipe space and that of water or steam is carried out in pipes. That design ensures the possibility of repairing the SG by plugging the separate pipe, which has lost its tightness, without dismantling the SG or breaking the primary circuit.

Shut-down regimes, the regimes of start-up and cooling down are realized in the easiest way if the SG operates by using multiple circulation in the water -steam circuit.

Both mechanical upright pumps with a turbine or electric driver and electromagnetic pumps may be used for coolant circulation.

During last period eight NSs with RIs using liquid metal coolant (LMC) have been constructed [11]. The first experimental NS of Project 645 had two reactors. Other seven NSs of Project 705 (according to the NATO terminology - "Alpha") had one reactor. Due to its high-speed qualities this NS was put down into the Guinness Book of Records. Besides that, two full-scale ground-based reactor test facilities were constructed and put into operation (the prototypes of NPIs at NSs at IPPE in Obninsk and at NITI in Sosnovy Bor). The total RI operating time added up to about 80 reactor-years.

3. THE CONCEPT OF USING REACTORS COOLED BY LBC INNUCLEAR POWER

3.1. The Fundamental Statements of Safety Concept

Priority for realizing the safety requirements in comparison with the other has become obvious after happening the Chernobyl accident.

An intensive development of NPP new concepts for the future has been launched all over the world, their safety would be more based on RI inherent self-protection allowing to eliminate deterministically the possibility of the severest accidents which result in catastrophic consequences not only in the event of failing the safety technical systems and personnel's errors but also in the event of ill-intentioned actions (terrorism, diversions, military attacks) and natural disasters.

Nowadays there are two approaches to ensuring the RI safety.

The first approach is traditional and is based on increasing the amount and efficiency of various protecting and localizing systems which reduce the possibility of severe accidents and hazardous factor of their results. Practical realization of this approach, theoretically based on probabalistic safety analysis (PSA), results in more complicated and expensive installation, deterioration of its other characteristics and, nevertheless, in principle, does not exclude the possibility of severe accident with catastrophic consequences because there have not been eliminated the internal causes of arising the accident.

The second approach is based on the concept of the RI with inherent safety ensuring its self-protection in which the causes of arising the severe accidents with catastrophic consequences have been deterministically eliminated by nature laws. This approach does not require the construction a lot of protecting and localizing systems, which, in some cases, themselves may become the causes of accidents.

There is no need in complex substantiations of safety either by using many calculative and experimental works in the frameworks of abstract scenarios for severe accidents in progress, by constracting expensive large-scale test facilities. Most consistently this approach has been developed in the USA by Prof. A. Weinberg [12], and in Russia by Prof. V.V. Orlov[13].

Neither the first approach nor the second one is realized in its pure form. Each of them possesses the elements of the other. Nevertheless, proceeding from the concept "SAFETY CULTURE" [14], when developing the advanced RI, it seems necessary to use the approach most based on the inherent safety.

It is due to the fact, that the large-scale advanced RI must meet the stricter safety requirements because the probability of severe accident specified in active standard documentation gives the socially acceptable expected value of frequency of its realization only for today's level of the NPP reactors service time, which is less than 10000 reactor-years.

If in the future this factor increases ten times, it will take us to reduce this probability ten times for 1 reactor-year in order to keep today's expected value of frequency of severe accident realization [15].

For traditional types of NPPs, which safety is usually substantiated by PSA methods, it takes us to increase the amount of protective barriers (double containment), localizing systems (corium trap), etc. The result of it will be inevitable going up the NPPs cost and losing their competitive ability in comparison with electric plants that use organic fuel. We are watching the symptoms of that process even now.

Inadequacy in substantiating NPPs safety for future NP by using PSA methods is the result of the fact that the PSA considers failures of technical devices and operational personnel's errors to be occasional events, which probability may be estimated only with high uncertainty. Low probability of the severe accident is neither the proof of its impossibility nor the evidence of the fact that it might happen not earlier than thousands or tens of thousands of years later. Besides that, if there were people's ill-intentioned actions, that should be taken into account, these events would not be occasional but programmed in advance. In that case the PSA conclusions lose their validity completely.

3.2. Substantiation of Choosing the Fast Reactor Cooled by LBC

Liquid metal cooled fast reactors are classified as RIs which safety is ensured principally due to their inherent self-protection. It is associated with a number of their internal features.

Absence of poisoning effects in the fast reactor (FR), low value of negative temperature reactivity coefficient, compensation of fuel burn-up and slagging processes by plutonium generation as well as partial reloadings enable to ensure the operative reactivity margin to be less than delayed neutron share and to diminish or eliminate the probability of runaway by prompt neutrons in the reactor under operation conditions.

LMC used for FR cooling motivates considerably constructional and thermal scheme of the RI, technical and economical characteristics of the NPP.

Among the LMCs used in NP sodium is the most commonly used.

Choosing this LMC for fast reactors was caused by its possibility of intensive heat removal due to its good thermal and physical properties. It enabled to provide short plutonium doubling time that was obligatory requirement at the early phases of designing the fast breeder reactors in the 60s and 70s years, and was caused by the fact of unproved forecast of very high NP development rate and, therefore, need for fuel self-providing. That was the reason why in the 50s Academician A.I.Leypunsky, who considered various LMC for cooling the FRs, preferred sodium, though LBC was initially considered for these purposes [16].

At present and in the foreseeable future there is no need for such short plutonium doubling time which can maintain the FR cooled by sodium. The necessity for designing fast breeder reactors has been postponed to many decades [17]. It enables us to return to the opportunity of using LBC for cooling the FR, one should take into account many-year experience we have gained by using this coolant in the NS reactors designed under the IPPE scientific supervision.

It should be highlighted that the experience of using sodium coolant has been gained under the conditions of industrial operation of power reactors at NPPs and could be used at once, whereas the experience of using LBC has been gained under the conditions of RI operation at the NS, which were different from those at NPP in scale factor, and requires applicable adaptation to new conditions. However, these circumstances should not be the reason for not using LBC in NP if it has weighty backgrounds.

These backgrounds are the following:

- Enhancing reactor safety due to elimination of coolant boiling (coolant boiling point is ~ 1670°C, boiling point of sodium is ~ 870°C) in the most heat stressed fuel subassemblies (FSA) even in the case of the severest accidents. It makes the realization of coolant positive void reactivity effect practically impossible. Besides that, the use of chemically inert LBC eliminates occurrence of explosions and fires if there is coolant contacting with air and water which is possible in emergency situations;
- Improving technical and economical parameters due to using two-circuit scheme of heat removal, eliminating some safety systems, systems of accident localization, simplifying the technology of managing the SNF;
- Solving the number of principal problems on RIs reliability and safety: ensuring the proper coolant quality and its maintenance in the course of operation, ensuring radiation safety associated with forming alfha-active polonium-210 radionuclide, etc. These tasks have been developed for the NS's RIs.

The expediency of using eutectic lead-bismuth alloy as primary circuit coolant in nuclear reactors is due to its physical and chemical and thermodynamical characteristics which enable to meet the safety requirements the most completely.

Extremely high boiling point about 1670°C allows low pressure in the primary circuit with the 400...500°C outlet reactor coolant temperature that ensures high steam parameters. Thus the RI structure is simplified and its reliability increases.

Natural properties of LBC such as high boiling point practically eliminate the possibility of the primary circuit overpressurization and reactor thermal explosion if coolant is accidentally overheated, because there is no pressure increasing in this case.

Impossibility of coolant boiling enhances the reliability of heat removal from the core and safety because there is no phenomenon of heat removal crisis.

Low pressure in the primary circuit enables to reduce the thickness of reactor vessel walls and to use for its fabricating less strong austenite steel being resistant to radiation embrittlement under operation conditions and eliminate the possibility of vessel brittle damage. This enhances safety and eliminates the restrictions on temperature change rate on conditions of thermocycling strength and radiation life time of reactor vessel.

Loss of coolant with its circulation interruption through the core if there is failure of the main reactor vessel tightness (postulated accident) with suggested integral design of the RI can be eliminated by introducing the safe-guard shell, small free volume between the main reactor vessel and the safe-guard shell and impossibility of coolant boiling-up off in the case of loss of tightness by primary circuit gas system.

In the case of failing all systems of emergency cooling down (postulated accident,) elimination of core melting down under heat decay effect and keeping the vessel of small and medium-sized power reactors intact is ensured completely by passive way with large margin to boiling due to accumulation of heat in internal reactor structures and in coolant with short-time increase of its temperature. In this case heat is removed through the reactor vessel (which temperature increases correspondingly) to the tank of water storage around the reactor vessel and after its boiling away (in the case of full switching off and interruption of cooling down systems work) into the ambient air by its natural circulation.

In the case of emergency overheating and simultaneous postulated failure of emergency protection systems (EPS) down to the level which does not cause the core damage the reactor power decrease is ensured by reactivity negative feed backs.

In the core and within the RI there are no materials which yield hydrogen as a result of thermal and radiation exposures and chemical reactions with coolant. The coolant itself reacts with water and air very slightly, their contact is possible if the circuit has lost its tightness. Thus, the possibility of arising chemical explosions and fires caused by internal reasons is eliminated completely.

Slight chemical activity of LBC enables to realize two-circuit scheme of heat removal, so the RIs become simpler and cheaper. The emergency processes associated with tightness loss by primary circuit pipes and intercircuit SG leaks occur without hydrogen yield and any fire and explosion hazards.

The elimination of water or steam penetration into the core (if the SG leakage is large) and consequent overpressurizing the reactor vessel designed for maximal possible pressure, if that accident occurs, are ensured by coolant circulation scheme. In this scheme steam bubbles and water drops are thrown out on the free coolant level by upgoing coolant flow. Thereby steam effective separation occurs in the gas space of the primary circuit above the coolant level, whence steam goes to the system of emergency condensers operating passively, and through the ruptural membranes it goes to bubbler if these condensers fail postulatedly.

The operation experience of the RI using LBC at the NS has illustrated the possibility of RI safe operation during some time under conditions of small SG leakage, which does not cause any significant deviations of the designed engineering parameters. This fact allows to realize necessary repair works not urgently but at the convenient time.

The coolant does not boil if the primary circuit loses its tightness, and has the property to retain iodine, as a rule, its radionuclides represent the major factor of radiation danger just after the accident as well as the other fission products (inert gases are exception) and actinides. This reduces sharply a scale of radiation consequences of that accident in comparison with pressurized water reactors.

The containment above reactor serves as the additional safety system barrier. Its main purpose is the protection against external effects. Low storage of potential energy in the primary circuit restricts the RI destruction scale to only external impacts.

Extremely high safety potential peculiar to this type of the RI is characterized by the fact that even when such initial events as containment destruction and primary circuit loss of tightness coincide, neither reactor runaway, nor explosion and fire occur, and the radioactivity yield is lower than that one which requires the population evacuation.

Taking into account that energy stored in the coolant (heating, chemical and compression potential energy) is minimal in comparison with other coolants used and previously mentioned physical special characteristics of fast reactors and RI integral design, one could look forward to designing the RI of ultimately achievable self-protection.

It makes possible and expedient the use of that type reactors not only for electric power generation at the NPP but also for simultaneous heat generation at the NHPP with their locating near large cities or for sea water desalination. On the base of such reactors NP would become not only socially acceptable for population but also socially attractive if it gained economical competitive ability with heat electric generation plants using organic fuel (we have all backgrounds for it).

The high level of self-protection of reactors considered makes expedient their use for nuclear transmuting NP long-lived radioactive waste.

3.3. The Problems of Increasing the Overall Bismuth Production and Using the Lead Coolant

The factor limiting LBC use in large-scale future NP depends on deficiency of today's bismuth manufacturing which has been determined by its consumption level.

Today's bismuth situation looks like uranium one during the period from 1906 to 1940 when only 4000 t of uranium were mined over the 34 years. But in 1980 the world's uranium mining (except the ex-USSR) reached 40000 t per year [18]. The explored uranium resources increased greatly too.

One should point out that bismuth content in the Earth's crust is one-fifth as high as that of lead [19]. However, deposits of bismuth with high content of about 5-25% are very rare and locate in Bolivia, Tasmania, Peru and Spain. That is why 90% of world bismuth has been manufactured from the wastes of lead-refining, copper-smelting and tinning plants.

The experience of the USA and Japan has revealed that equipping copper-smelting, tungsten and lead plants with dust-catching and dust-utilizing systems can essentially enhance bismuth extraction and has great significance for the environment.

If the problem of developing the future large-scale NP with that type of reactors is solved positively, it is necessary to carry out bismuth geology works that have not been performed in necessary scale until today because of lacking demands.

In Russia that work can be launched at MINATOM enterprises being the members of AO "Atomredmetzoloto". According to VNIPI Promtechnologia information, the production of bismuth of about 2500-3000 tons per year, together with gold and other metals, may be organized in the south-east of Chita Region where the resources of gold-bismuth ore have already been explored. That volume of bismuth production will ensure the year input of 2,5-3,0 GWe for the NPP using RI SVBR-75 and 1 GWe for the NPP using SVBR-600 (if aggregate power of the RI increases, specific expenditure of bismuth increases too).

The cost of bismuth is ten times as much as that of lead and is a very small part of capital costs for NPP construction. It should be taken into account, that coolant is not spent and could be used again in other RIs.

RDIPE proposes to consider lead coolant as an alternative to lead-bismuth alloy because the scales of lead production do not limit the rate of large-scale NP development.

However, use of lead coolant is associated with some engineering problems. Due to the higher lead melting point (it is 327°C in comparison with 125°C for eutectic coolant), lead coolant temperature must be increased significantly. It complicates the solution to the problem on coolant technology, structure materials corrosion and mass transfer. Being applied to the lead-bismuth coolant, the problem has taken about 15 years for its solving. It will demand changes from existing steam parameters to overcritical ones which have not been used at the NPP. Besides that, it results in more complicated RI operation because of great possibility of forming solid "sows" in the primary circuit under transitive regimes, accidents, repairs, refueling. Nowadays the works on mastering the lead coolant are at their initial phase.

Taking into account all mentioned above, use of lead coolant would be justified only if the rates of power capacities increase for the NPPs with the RIs considered were high enough, and expenditures for increasing the annual bismuth production and its cost were put up as much that the increase of specific capital costs of NPP construction would not be economically reasonable.

It should be highlighted that there is no sharp boundary between lead-bismuth eutectic alloy and pure lead. As bismuth has been in deficiency, one can consider non-eutectic alloy with bismuth content decreased up to 10% (versus 56% in eutectic alloy). Being compared with lead coolant, its melting temperature is decreased by 77°C (to 250°C) and that facilitates RI operation and reduces maximal temperatures of fuel elements claddings up to the values tested for eutectic coolant under the conditions of long-term operation tests.

Mastering the technology of alloy with 10% of bismuth content further ensures, if necessary, gradually to introduce lead coolant use.

4.PROPOSALS ON USING LEAD-BISMUTH COOLANT

4.1. The Near-Term Prospect up to the year 2030 (On the Basis of Multipurposed Reactor Module SVBR-75/100)

Currently the NP development meets significant difficulties in many countries. First of all it relates to complicating and cost rising of NPPs due to essential enhancing the safety requirements.

Increasing the unit power of the nuclear power units with light water reactors aimed at specific capital costs reducing enhances the total cost of the construction and the constructing term. This is unfavorable for receiving a credit.

The possibility and expediency of developing the NP based on unified small power reactor modules SVBR-75/100 with fast neutron reactors cooled by lead-bismuth eutectic coolant (LBC) is substantiated for the nearest decades in the paper.

The properties of self-protecting the RIs using LBC allow to create ecologically pure nuclear power sources of enhanced safety.

For the small power RIs using LBC long-term passive heat removal through the reactor vessel into the environment is ensured without core damage in postulated accident with complete failure of heat removal and emergency protection systems.

Characteristic features of small nuclear power plants (SNPP) are

– transportable or modular-transportable design if SNPP is assembled by using completed units with possibility of their transporting by any kind of transport;

– improved (in comparison with RIs of other types) weight and overall dimensions parameters of SNPP and its separate units;

– long work without refueling (about ten years) that enables not to reload fuel at the site of SNPP operation if there is no suitable infrastructure (this possibility is realized in the RI with the power lower than 100 MWe), and reduces the risk of unauthorized plutonium proliferation;

– the possibility of SNPP reactor unit transportation in order to reload fuel and, after serving its term, to the central base of technical service under the conditions of nuclear and radiation safety and additional physical protection (coolant and fuel assemblies "freezing" in the reactor);

– economical competitive ability with other types of SNPP.

The project of that SNPP "Angstrem") gained the first place among its power group in the SNPP project competition held by Nuclear Society [20].

The closeness of scale factors of the RI at SNPP and the NPI at the NS ensures continuity of main engineering solutions and enables creation of SNPP in very short terms if there is a certain customer. SNPPs of the type considered satisfy the principle which is the most acceptable for the customer: "I build, own, operate, decommission up to the "green" lawn".

The great interest to SNPP of 50-150 MWe power was revealed at the IAEA working group meeting on using ship reactor technologies for civilian needs (July 1998, Obninsk, Russia) such as producing electric energy, desalinated water and heat generation [21

It is followed from the USA Report, in which demand for such reactors is valued to be 50-70 after 2015. The USA suggests to organize the international cooperation on realizing the design that satisfies these requirements the most completely.

The analysis of unique experience of operating the NPIs using LBC, that has been performed by Russian experts, has testified that in developing small and medium-sized power reactors for civilian NP needs the causes of the accidents happened and the operation difficulties have been eliminated.

The designs considered have shown that NPPs with reactors cooled by LBC satisfy the modern safety requirements, non-proliferation of Pu and can be economically competitive with traditional NPPs. For those NPPs entering the market of nuclear technologies (they could be attractive for developing countries and for economically effective replacement of removing NPP's powers capacities) there is need for construction and operation of that reactor in Russia. Now that perspective is been studied.

On the basis of the concept presented and the experience gained during the last years under the scientific supervision of SSC RF IPPE named after Academician A.I.Leypunsky a number of proposals for civilian small and medium-sized NPPs have been developed at EDO "Gidropress: renovating of NPP units which operation term has been exhausted; regional nuclear heat power plants (NHPP) of 100-300 MWe power which need near cities location; large power modular NPPs (~ 1000 MWe) like US concept PRISM or Japanese concept 4S; nuclear power complexes for sea water desalinating in developing countries which meet non-proliferation requirements, reactors for Pu utilization and minor actinides transmutation.

Their brief description is presented below.

4.1.1.Multipurposed Reactor Module SVBR-75/100

RI SVBR-75/100 is designed for generating steam which parameters enable to use it as working medium in thermodynamical cycle of turbogenerator installations. It is possible to vary the steam parameters according to the needs. Today the base variant of RI SVBR-75 [22] has been developed for generating the saturated steam under the pressure of 3,24 MPa, i.e. the pressure which is produced by the SG of the Novovoronezh NPP (NVNPP) 2-nd unit, and when turbogenerator with intermediate steam superheating is used, this enables to generate electric power of about 75 MWe when working under condensation regime.

The design of RI SVBR-75 have two-circuit scheme of LBC heat removal for the primary circuit and steam-water for the secondary circuit. The integral design of the pool type is used for the RI primary circuit (see Fig. 2). It enables to mount the primary circuit equipment inside the one vessel. RI SVBR-75 includes the removable part with the core (the reactor itself), 12 SG modules with compulsory circulation over the primary circuit and natural circulation over the secondary circuit, 2 main circulation pumps (MCP) for LBC circulation over the primary circuit, devices for controlling the LBC quality, the in-vessel radiation protection system and buffer reservoir which are the parts of the main circulation circuit (MCC).

The scheme of coolant moving within the MCC is as follows: through the windows of reactor outlet chamber the coolant heated in the core flows to the inlet of the SG twelve modules which have parallel connection. It flows from top to bottom in the intertube gap of the SG modules and is cooled there. Then the coolant penetrates into the intermediate chamber, from which it moves in the channels of in-vessel radiation protection system, cooling it, to the monoblock upper part and there it forms the free level of "cold" coolant (peripheral buffer chamber), further from the monoblock upper part the coolant flow moves to the MCP suction inlet.

The adopted circulation scheme with free levels of LBC existing in the monoblock upper part and SG module channels, which contact the gas medium, ensures the reliable separation of steam-water mixture out of coolant flow when the accidental tightness loss of SG tube system occurs, and existing of gas medium ensures the possibility of coolant's temperature changes.

Monoblock is placed in the tank and is mounted there (see Fig. 3). The tank is filled by water and is designed for cooling the RI in case of beyond design accidents. The gap between the major vessel and safe-guarding one is chosen to ensure the circulation circuit disrupture in case of accidents related to the tightness loss by the monoblock major vessel.

The secondary system is designed to operate the steam generator producing saturated steam with multiple natural circulation through the evaporator-separator, as well as to provide the scheduled and emergency RI cooling by using steam generator.

The design provides three systems of heat removal to the heat sink both in scheduled and emergency reactor core cooling. First system includes normal operation RI and turbine generator

Реакторный модуль СВБР-75.
Reactor plant module SVBR-75.

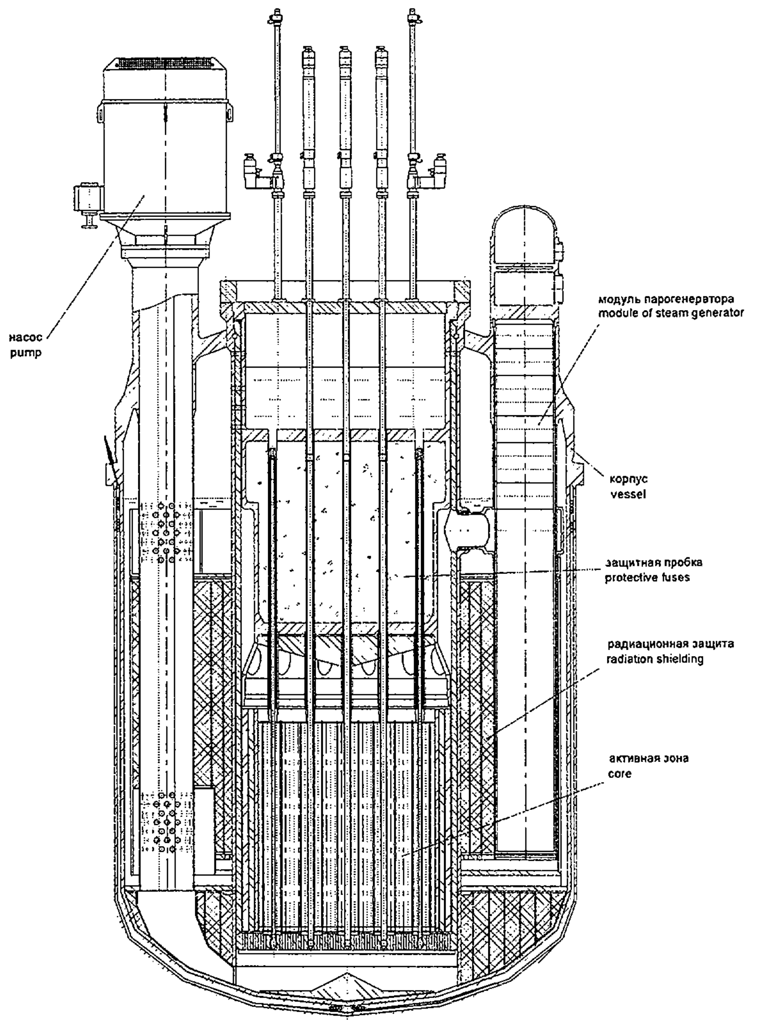


Fig. 2

equipment and systems. The system is cooled by the primary heat removal via steam generator heat exchange surfaces, steam being dumped to the turbine generator systems (TGS).

The second heat removal system is an independent cooling system (ICS), which includes, besides a part of primary and secondary circuit equipment, a loop separator-cooling condenser with natural circulation. Via this loop the heat is removed to the intermediate circuit water. This system ensures independent (from the turbine generator systems) reactor cooling and independent reactor plant operation at a constant power level up to 6 % N_{nom} at the nominal steam pressure. In case of total RI de-energizig the system ensures cooling of the reactor over several days. Connection/disconnection of ICS is realized with no operator action and without using external power supply systems.

Third core heat removal system is a passive heat removal system (PHRS). The heat is removed from the monoblock to the water storage tank located around the monoblock vessel. This system ensures the reactor core cooling in case of postulated maximal accident with all secondary equipment failed, reactor protection system failure and total de-energizing of the NPP.

The principal technical parameters of RI SVBR-75 are presented in Table 1.

RI SVBR-75 operates for eight years without core refueling. During this period there is no need in carrying out fuel works. At the initial stage the use of mastered oxide uranium fuel in the uranium

Реакторная установка СВБР-75.
Reactor plant SVBR-75.

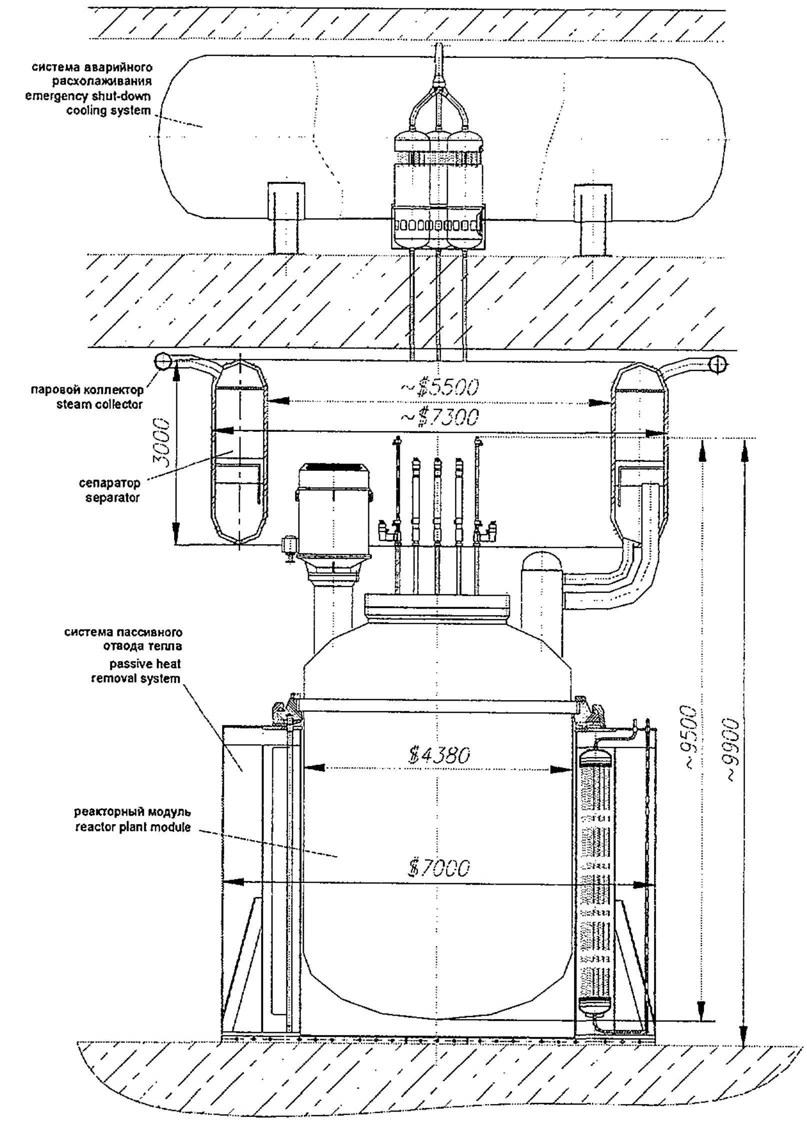


Fig. 3

closed fuel cycle is provided similar to that in reactor BN-600. Further the use of dense uranium and plutonium nitride fuel is possible. In this case the core breeding coefficient is more than 1 and in the plutonium closed fuel cycle the reactor would operate by using only depleted waste pile uranium.

The refueling is performed after lifetime ending. Refueling means the complex of works on restoring the reactor full power resources which includes core replacing works, as well as decommissioning and commissioning works associated with them. Cassette by cassette fuel unloading out of the reactor vessel is provided and loading of the fresh core as a part of new removable part is provided as well. Core refueling is realized by using special refueling equipment. Unloaded FSAs are placed in special penals with liquid lead which solidifies further.

RI SVBR-75/100 is designed on the design base of RI SVBR-75 and distinguishes from it only by SG operating in one through regime and generating the superheated ste°am of 400°C temperature and 9 MPa pressure. Thus the electric power is ensured to be of about 100 MWe.

TABLE 1. PRINCIPAL TECHNICAL PARAMETERS OF RI SVBR-75

Parameter	Value
Number of reactors	1
Rated heat power, MW	268
Electric power, MWe	75
Steam production rate t/h	About 487
Steam parameters:	
-Pressure, MPa	3,24
-Temperature	238
Feed water temperature, °C	192
Primary coolant flow rate, kg/s	11180
Primary coolant temperature, °C	
-core outlet	439
-core inlet	275
Core dimensions, DxH, m	1,65x0,9
Average value of core power stress, kW/dm^3	135
Average value of fuel element stress, kW/m	~ 22
Fuel:	
-type	UO_2
-U-235 mass loading, kg	1476
-Average U-235 enrichment, %	15,6
Steam generator (SG) numbers	2
Evaporator numbers in SG	6
Evaporator dimensions DxH, m	~ 0,6x4
Numbers of PCMPs	2
PCMP electric driver power rate, kW	400
PCMP head, MPa	~ 0,5
Primary circuit coolant volume, m^3	18
Monounit vessel dimensions, DxH, m	4,53x6,92
Designed earthquake	Of magnitude 9 (MSK)
Designed construction terms (months)	36

4.1.2. Renovation of NPP's Units with Exhausted Lifetime

The number of NPP's units which lifetime has been exhausted is growing in the NP of many countries. It needs for huge expenditures on withdrawing the units out of operation and constructing the new ones for replacing the removing power capacities. At the same time there is an opportunity for untraditional solving this task by using NPP's units renovating. Renovating means replacing the Risfor units with exhausted lifetime by new Ris, using the NPP's existing buildings and structures with full replacing of removed power capacity. However, this way of replacing removed power capacities rigidly restricts the type of the RI used for renovation:

- the possibility of mounting the RI in existing rooms after dismantling the equipment of the "old" RI (the reactor which dismantling is followed by great radiation doses is an exception);
- satisfying the regulation requirements for safety including "old" units without containment.

That way of replacing the removing power capacities of the 2-nd, 3-rd and 4-th units of NVNPP based on the unique Russian lead-bismuth technology has been suggested by SSC RF IPPE together with EDO "Gidropress" and GNIPKII Atomenergoproekt and is been developed according to the task of concern "Rosenergoatom".

SVBR-75 nominal power is chosen to be 75 MWe due to limited dimensions of NVNPP renovation units' SG compartments which do not enable to install the large power module, necessity for ensuring the equality of generated steam and feeding water consumption for SVBR-75 and RI

VVER-440 SG, possibility of reactor module complete plant fabrication and its transportation by the railway, as well as closeness of the scale factor to NS's RIs that enables to use some developed technical solutions and reduce R&D. For replacing the power capacities of the 2-nd unit four SVBR-75 modules are installed in SG compartments, and six modules are installed for each of 3-rd and 4-th units.

Arrangement of modules in SG compartments of the MCP of NVNPP's 2-nd unit is presented in Fig. 4.

The great economical efficiency of renovating the "old" units is expected: specific capital renovation cost is 560 dollars per kWe, that is half as many as that for constructing the new NPP block[23]

4.1.3. Regional NHPP Based on Module R-75/100

In the world many regions and first of all medium-sized and large cities face with serious difficulties in energy supplying especially heat supplying in winter.

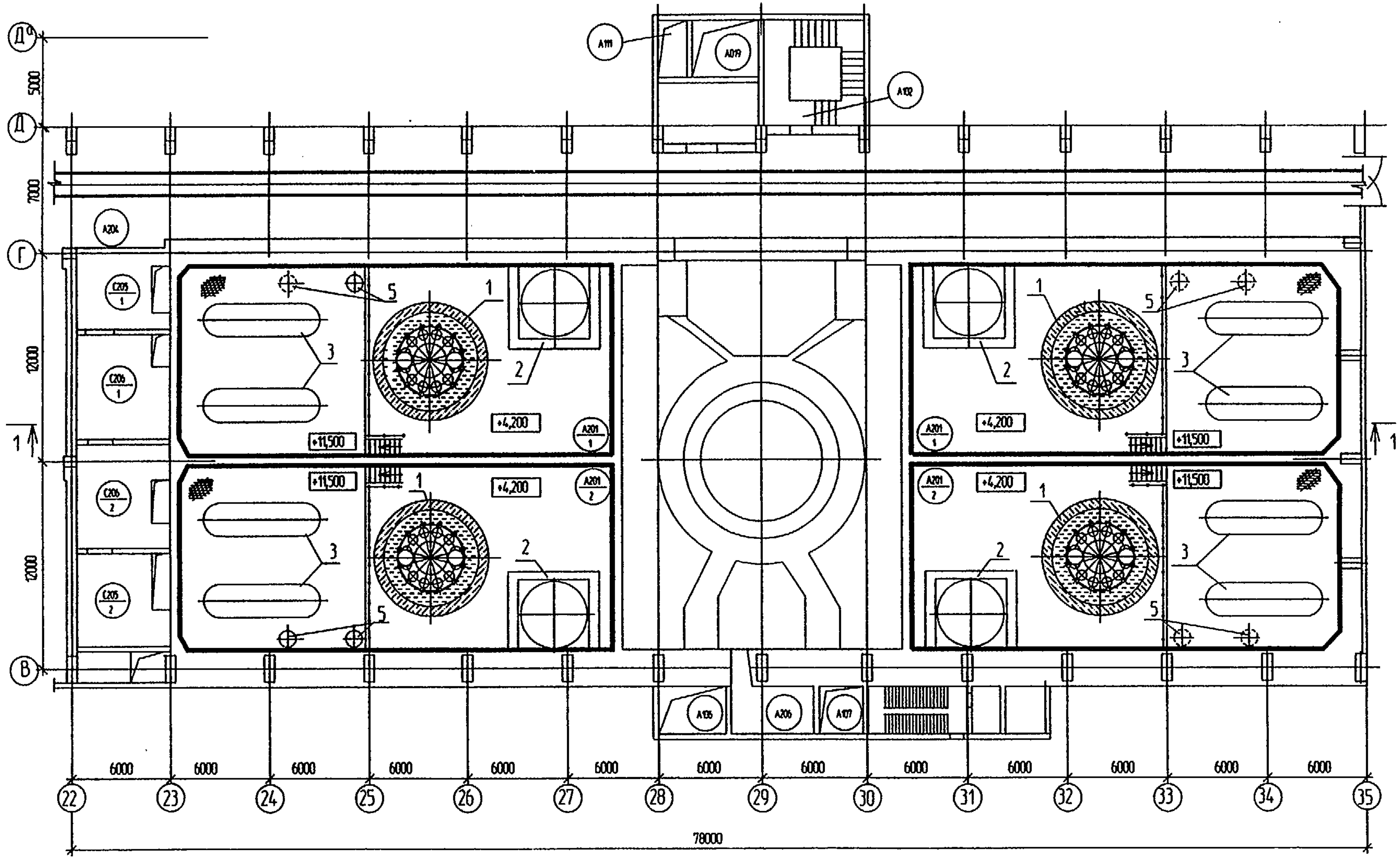
One way for solving this problem is use of nuclear heat power plants (NHPP). However, use of traditional type RIs (which use water under the high pressure for reactor cooling) for these purposes needs designing the number of additional safety systems if compared with those accepted for NPPs situated at the distance of 25 km or more from the cities. This results in going up the NHPP cost and at the same time does not eliminate the principal possibility of scarcely probable nuclear accident with severe consequences because the high pressure in the reactor, which is the internal cause of its arising, is not eliminated.

The high level of SVBR-75/100 reactor module inherent safety makes it possible and expedient to use it simultaneously for producing electric energy at the NPP and for heat generating at the NHPP which needs near city's location. So we eliminate the possibility of arising severe accidents accompanied by explosions, fires with unpermitable radioactivity exhausts which require population evacuation beyond the NHPP site not only if there are personnel's errors and equipment failures but if they coincide, if there are terrorist groups' actions. We can construct NHPPs of 100-300 MWe by using these standard modules.

For the major vessel of regional NHPP unit for RI SVBR-75/100 the principal design solutions distinguish from those for traditional type reactors. Small dimensions of monoblock SVBR-75/100 and developed properties of inherent safety of FRs with LBC require the RI protection from only those external effects: aircraft falling, shock waves, maximal computated earthquake (MCE). There is no need for designing the tight shell withstanding significant internal pressure. Small dimensions of protected reactor compartment and simple scheme of RI enable to reduce the terms of NHPP unit construction and significantly reduce the construction cost.

The simplicity of automated control system (ACS) conditioned by using passive systems for cooling down the RI facilitates the construction cost reducing.

As it has been assessed by experts, the economical competitive ability with HPP using fossil fuel will be ensured due to the following facts: almost lack of expenditures on nuclear fuel transportation; long lifetime of the fast reactor core, which ensures without refueling RI operation during about 8 years; low cost of spent nuclear fuel storing; almost lack of liquid radioactive waste and expenditures for its recycling; complete plant fabricating of the RI and possibility of its transportation by car, railway or sea to the NHPP constructing site that reduces the constructing terms and approaches them to those of traditional HPPs and reduces the investment cycle; sharp reduction of expenditures for designing safety ensuring systems due to RIs' high inherent safety; high commercial production due to great demand for these NHPPs; possibility of export delivery of these RIs. The cost of these NHPPs is one-fifth - one-tenth as many as that of large NPPs.



Plane on the point 4.200

ПЛАН НА ОТМЕТКЕ 4,200

Figure 4

4.1.4. Large Power Modular NPP

On the basis of commercially produced modules SVBR-75/100 it is expedient to develop the design of modular NPP of large power (1 GW and more at the same unit). The prospect for that principle of designing NPP is shown in conceptual design developed in the USA (PRISM) [24] and in Japan (4S) [25]. However, use of this principle for LBC cooled reactors is the most effective.

The economical gain is achieved due to: constructing volumes reduction because of eliminating the number of safety systems and localizing systems, reducing the specific material expenditure (including bismuth demands) as compared to traditional reactors of large power, reducing the fabricating cost due to high commercial production, reducing the NPP constructing terms when reactor modules are delivered to the constructing site in high plant readiness. It enables to improve the conditions of credit receiving and repayment and to increase the competitive ability of NPP. The preliminary assessments have revealed that the capital cost of constructing such NPP is expected to be not more than that for constructing the NPP's VVER-1000 unit.

4.1.5. Dual-purpose nuclear desalinating power complex for developing countries

Many developing countries in Africa an Asia suffer from deficiency of fresh water and electric energy. The majority of these countries do not have sufficient own resources of fossil fuel, which can meet their demands. In some countries fuel transporting is difficult, there are no powerful electric power transmission lines. The marketing researches conducted recently by IAEA [26] have revealed that in many cases nuclear power sources of 100 MWe small power can be used economically effective for these purposes.

However, the developing countries' particularities concerned with not sufficiently high level of education, technical culture, social and economical development, as well as possibilities of arising local military conflicts, put forward special requirements to the nuclear power technology which are stricter than those for developed countries.

First of all, these requirements are RI inherent safety against severe accidents that is based on RI inherent properties ensuring safety not only in cases of personnel's errors and multiple failures of technical systems coincidences, but in cases of sabotage terrorist actions, etc. Besides, they must meet strict non-proliferation requirements [21], including that refueling in the country-user must be eliminated and due to this fact the lifetime duration must be 10 years or more. The opportunity for reactor unit transportation to the country-manufacturer in the state of nuclear and radiation safety for refueling and then transporting it to the country-user again must be ensured. Thefts of fuel must be technically eliminated as well. Besides, the competitive ability to the alternative resources of receiving fresh water and electric energy must be ensured.

RI SVBR-75/100 meets these requirements the most completely. It has extremely high safety potential, lifetime duration needed, ensures the regime of non-proliferation due to the following:

- use of uranium with enrichment less than 20%,
- lack of refueling in the country-user,
- opportunity of transporting the reactor module after ending its lifetime to the country-manufacturer in the state of nuclear and radiation safety with LBC "frozen" in the reactor.

4.1.6. Plutonium Utilization and Long-lived Minor Actinides Transmutation

In different countries the policy on managing plutonium, which quantity increases steadily, is different. It is determined by the fact that on the one hand Pu is very valuable fuel for the future NP using FRs. On the other hand it can be used for political and military purposes. Besides, Pu is high radiotoxic material and is considered to be NP dangerous radioactivity waste. Long-lived minor actinides belong to it too.

For the first case the issue of Pu managing (both extracted one and that is contained in spent nuclear fuel) results in long reliably controlled storing.

The second case results in the task of Pu transmuting into the form that reduces the risk of its unauthorized proliferation or its complete burning up and minor actinides nuclear transmutation.

For solving this task the works on mastering the new nuclear technology using accelerator-driven systems are carried out. The main stimulus for developing this technology is Pu and minor actinides blanket subcriticality that eliminates the prompt neutrons runaway nuclear accident.

Along with it, this task can be solved on the basis of already mastered technology. For example, during eight years one reactor module SVBR-75/100 can transmute about 1000 kg of Pu (weapon or reactor one) into the form protected against unauthorized proliferation ("spent fuel standard") at reducing its quality as a weapon material compared to the weapon Pu. In terms of 1 GWe - year 1,25 tons of Pu will be utilized in those reactors. If minor actinides (first of all amerithium) is introduced into fuel, their transmutation into short-lived radioactive wastes will take place.

The safety level needed will be ensured due to developed propertied of RI SVBR-75/100 inherent safety which have been mentioned above.

5. THE DISTANT FUTURE TASK (AFTER THE YEAR 2030) LONG-TERM NP DEVELOPMENT IN CONDITIONS OF LIMITED NATURE URANIUM RESOURCES

The development of power and NP especially is very sluggish. That is why many countries, where there is developing NP, consider conditions under which NP can exist and develop in the middle of the next century. As far as Russia conditions are concerned, this time period would be characterized by the fact that available nature and enriched uranium resources which enable to develop NP and ensure export supplies without large costs for uranium mining and enriching would be exhausted. It would result into the necessity of searching, mining and developing new uranium deposits at hard-reached areas or into the orientation towards scale uranium import that would be the cause of increasing the fuel component of the cost of electric power generated by NPPs with thermal neutrons reactors.

The problem of NP fuel supplying could be solved by FR operating in closed nuclear fuel cycle (NFC) and enabling to involve effectively uranium-238 into power generation.

However, nuclear fuel-power complex development on this base inevitably yields to increasing specific capital costs and duration of investment cycle in comparison with modern and promising HPP using organic fuel. Under the market economy conditions when construction should be performed on the base of credits repayment it can deteriorate the NP competitive ability if consumers have free access to the wholesale electric energy and power market. Now this situation is forming in the USA, where constructed NPPs with light water reactors (LWR) is not putting into operation because of their low profits, even under conditions of the absence of the costs for fuel recycling and fuel elements refabricating. The situation is not going to improve even if weapon plutonium is involved into NFC with not complete fuel cycle closing (the absence of costs for reprocessing, minimal radiation characteristics during fabricating).

At the next century in the course of NP development based on the evolutionary improvement of traditional reactors the situation would be deteriorated because in order to keep up the value of severe accident risk at the today's existing socially acceptable level, the probability of that accident should be reduced inversely proportional to the number of reactor-years generated by the NPP units. It would inevitably yield to increasing the capital costs for increasing the NPP safety.

NP situation may be improved in more distant future when the resources of cheap natural gas have to be exhausted (the share of natural gas is the dominant one in electric energy production), quotas on releasing the greenhouse gases, significant increasing the cost of electric energy produced by HPP, controlling the environment have to be introduced.

There might be the long time interval (50-100 years) between the moment when NP loses its competitive ability after exhausting cheap uranium resources and that one when NP with closed NFC can be assuredly competed with that using organic fuel. Then NP would be less economically effective in comparison with power industry using organic fuel ("gas pause").

The circumstances considered make it actual to search for the concept of NP development during the "gas pause" period, which enables to overcome arising difficulties. One of possible concepts has been designed by SSC RF IPPE [27]. The main concept goal is to increase sharply the efficiency of natural uranium energy potential utilization without radio-chemical fuel reprocessing and fuel cycle closing for "gas pause" elimination.

5.1. Fast Reactor Operation in the Open NFC (Physical Concept)

FRs are commonly dealt with in the closed NFC in which they operate under the breeder regime. And plutonium is mainly accumulated in zones of its reprocessing from depleted uranium. The breeding ratio is less than that one in the core. The core make-up under partial refuelings is performed by fuel with fissile material content as that was in start-up load fuel. For providing core reactor make-up by built-up plutonium, closing the NFC with fuel reprocessing and fuel elements refabricating by exploitation of released plutonium is needed.

However, FR operation may be performed in non-traditional regime, its realization would enable to close the "gas pause". It is the FR with so called "without chemistry processing" fuel cycle that was first theoretically considered by S.M. Feynberg and E.P. Kunegin ("Kurchatov Institute", Russia) in 1958 [28] and then by K.Fucks and H.Hessel (Germany) in 1961 [29].

The fast reactor operation under that regime is not in conformity with the existing view points on the FR role in NP, as in this case built-up plutonium is neither extracted nor reused, but mainly utilized directly inside the reactor. This is the reason for reconsideration of traditional FRs.

In the reactor under consideration the start-up load of core consisting of fuel subassemblies (FSA) with comparatively high enriched uranium fuel (10...12%) is realized only once. (Certainly, there may be used U-Pu fuel containing weapon plutonium if the problem of its utilization is actual or yielded energetic plutonium). In the course of reactor operation the FSA of start-up load under partial refuelings are being replaced gradually by previously loaded make-up FSA where plutonium has been already built up, and their place is occupied by fresh make-up FSA with depleted or slightly-enriched uranium. There are no separate breeding blankets in that reactor.

The principal condition of ensuring reactor operation under that regime is core breeding ratio (CBR) being more than one. It enables the criticality to be maintained due to plutonium built-up. Thus, on the one hand, it is necessary the average plutonium concentration in the core to be more than critical one. On the other hand, this concentration should not exceed the equilibrium value, which is determined by the equality of the rate of forming plutonium out of uranium-238 and that of its burning-up. The reactor parameters and the regime of partial refuelings should be selected in such a way that the reactivity loss after partial refueling should be compensated by the reactivity increase during the operation period between refuelings. This reactivity increase is compensated by introducing the absorbing rods into the core. After partial refueling the reactor must be critical with the absorbing rods extracted.

Carrying out these conditions must meet the number of reactor requirements. The core must be of large dimensions. The volume share of fuel and its uranium density should be high. The share of FSA reloaded per one cycle of partial refuelings must be low (high refueling multiplicity).

The chosen scheme of partial FSA refuelings and reshufflings should also provide smoothing the energy yield distribution in the core. The problem is complicated because simultaneously in the core there are both FSA with large plutonium content that is close to equilibrium one and fresh make-up FSA without plutonium with very low energy yield. That is why zones with high and low plutonium content must be alternated.

After unloading the last FSA of start-up load out of the core the reactor goes into the self-maintaining operation regime. Thus criticality is maintained mainly by own plutonium and reactor uses only low-enriched uranium (LEU) or that from the waste pile (UWP). In this case the efficiency of using energy potential of natural uranium increases several times in comparison with LWR (see below).

5.2. Efficiency of Natural Uranium Energy Potential Utilization

Efficiency of natural uranium energy potential utilization (EUU) is determined as the ratio of the power generated by the reactor over the define time period (fission products mass) to the mass of natural uranium (NU) used during this period to provide the reactor operation.

For LWR this value is about 0.5% that corresponds to the consumption of NU of about 200 tons per one ton of fission products (or per 1 GWe - year).

For considered type of FR EUU increases as there is the number increase of fuel campaigns burnt up in the reactor. It is concerned with the fact that contribution of start-up load, which fabrication requires much NU, into the total reactor energy-generation decreases with increasing the number of fuel campaigns burnt up in the reactor.

If the use of UWP is ensured as make-up fuel, thus EUU would be maximal, achieving under the large number of campaigns the value which is equal to fuel burn up depth (% of h.a.). It corresponds to EUU increasing 10-20 times in comparison with LWR and is explained by the fact that for fabricating make-up fuel there is no need for NU. To ensure such EUU increase the fuel elements must meet the most stringent requirements for campaign duration, depth of fuel burning-up and the value of fast neutrons damaging dose.

If there is the increase of uranium enrichment in the make-up fuel, there would be the decrease of requirements for fuel elements operation conditions (see Table 2). But thus EUU decreases, being still several times as high as this one for the VVER-1000 reactor.

TABLE 2.THE DEPTH OF FUEL BURNING UP, FAST NEUTRONS DAMAGING DOSE, MICROCAMPAIGN AND CAMPAIGN DURATION AS THE FUNCTIONS OF MAKE-UP FUEL ENRICHMENT

X_5	g	D	Òìê	Tê
0	20,4	434	575	41
1	18,3	387	520	37
2	16,2	342	463	33
3	14,2	296	407	29
4	12,0	250	351	25
5	9,8	200	281	20

X_5 - make-up fuel enrichment, %
g - the depth of fuel burning up, % h.à.
D - fast neutrons damaging dose, dpa,
$\grave{O}_{mc}$ - the microcampaign duration, eff. days
T_c - the fuel campaign duration, eff. years

We can see it from Table 3 where the ratio of EUU for the SVBR-600 reactor with equivalent electric power of 625 MWe which has been considered as an example of realizing the FR operating in the open NFC, to EUU for VVER-1000 reactor has been presented. This ratio demonstrates the increase of the functioning time for NP using SVBR-600 reactors in comparison with that using

VVER-1000 ones in the open NFC under the same NPP's total power maintained and NU resources. If the enrichment of make-up fuel is taken to be 4,4%, as it concerns the VVER-1000 reactor, then EUU for the SVBR-600 reactor would be three times of that for the VVER-1000 reactor even in the fourth campaign. As a result, the consumption of natural uranium would decrease three times, i.e. the possible term of existing the open NFC would increase three times.

TABLE 3. THE COMPARATIVE EFFICIENCY OF NATURAL URANIUM ENERGY POTENTIAL UTILIZATION. (THE INCREASE OF OPERATION TIME FOR NP USING SVBR-600 REACTORS IN THE OPEN NFC IN COMPARISON WITH VVER-1000 UNDER THE SAME POWER AND NATURE URANIUM RESOURCES)

X_5 / n	0.2	1	2	3	4	5
1	1.05	1.00	0.96	0.90	0.87	0.84
2	3.15	2.79	2.45	2.19	1.98	1.81
3	5.24	4.33	3.57	3.04	2.65	2.35
4	7.34	5.66	4.44	3.64	3.09	2.69
5	9.44	6.85	5.14	4.11	3.42	2.93
10	19.90	11.15	7.22	5.33	4.23	3.51

X_5 - make-up fuel enrichment, %
n - a number of campaigns
EUU for VVER-1000 is considered to be 0,5%

Taking into account these results, it is expedient to use the flexible regime of reactor operation. In the first campaign the value of uranium enrichment in the make-up fuel is chosen in such a way that the fuel elements operation conditions are to be within the frameworks of the verified values. But the EUU will not be the best.

In the next campaigns, as the experience is being gained, the uranium enrichment in the make-up fuel will be decreasing, and the increase of EUU will be the result of it.

To achieve the maximal possible EUU there should be provided the possibility on exhausting the RI service term to use its fuel load, in which plutonium has been built up in quantities ensuring the reactor criticality, as the start-up load for the new RI.

The significant increase of EUU in comparison with LWR ensures the possibility of long-lived existence of the open NFC under the limited NU resources. It enables to postpone the necessity of introducing large-scale reprocessing the spent nuclear fuel (SNF) over 50 to 100 years, to save much money and eliminate the risk of nuclear, radiation and ecological accidents possible for this period if corresponding enterprises are functioning. By assessments to realize this possibility under the limited cheap uranium resources phasing out the LWR and introducing the FR of the SVBR-600 type into NP might be starting in 2030.

Nevertheless, as in the long-term prospect the SNF reprocessing and plutonium recycling are necessary, the search and development of the most economically available, safe and ecologically pure processes of SNF reprocessing must be carried out within the required scales.

5.3 Managing the Spent Nuclear Fuel

When there is long NP functioning in the open NFC the workload for repositories with burnt up FSA increases. These repositories must be of suitable volumes and provide long safe and controlled storage of the spent FSA. Enhanced safety of burnt up FSA storage can be reliably ensured because there are some barriers (fuel matrix, fuel element cladding) at the reactor for radioactive products release from FSA into the environment, and the additional ones can be designed (e.g., the capsule with burnt up FSA is filled by liquid lead and then "frozen"). If the spent fuel is stored in the "dry" repositories, it is not exposed to any attacks resulting in damage of protective barriers. Comparatively low volume of the spent FSA per 1 GW(e)-year, in comparison with VVER-1000, caused by 3 or 4 times deeper fuel burning up, facilitates the designing reliable physical protection resistant to external attacks and reduces the repository cost.

In addition to it, the reactor operation in the open NFC almost eliminates the possibility of unauthorized plutonium proliferation and utilization of built-up plutonium for the war purposes because it exists in the spent fuel together with high-radioactive fission products ("spent fuel standard").

It will be promoting to the rise of political stability in the world. It should be also highlighted that the risk of unauthorized plutonium prolifiration out of the SNF repositories after accumulating the threshold plutonium quantity making the fabrication of several hundreds of nuclear charges available does not increase proportionally to the total quantity of plutonium in the repository.

Turning over the spent nuclear fuel repositories (SNFR) to the International Guarantees facilitates to reducing the risk of plutonium thefts.

5.4. The Key Results of the Investigations Performed

Results of the investigations carried out have corroborated the feasibility of FR operation with make-up under partial refuelings by slightly enriched or depleted uranium. In this case the EUU highest value is achieved if the core dimensions are: (D x H$\cong$ 4.0 m x 1.4 m), fuel volumetric fraction is not lower than 60%, there is utilization of fuel with high uranium density (~11 g/cm^3) and formation of possibly harder neutron spectrum in the core (there is lack of light nuclei).

The formation of harder neutron spectrum has been influenced by LBC using, thus the higher value of CBR has been provided, which has had a key part in reaching the EUU biggest value. The calculations have demonstrated that for sodium coolant, which can moderate neutrons more effectively, it is hard to provide the reactor criticality if it has been made up by depleted uranium.

In the case of using metal alloyed (10% of Zr) uranium fuel with 75% effective density of theoretical one, the reactor can utilize waste uranium as the make-up. Thus the highest EUU is ensured (about 20%). In this case the burn-up depth achieves about 20% of h.a. (that is justified at experimental assemblies of EBR-2 reactor), fast neutron damaging dose on the fuel element cladding material accounts for approximately 430 dpa (it is twice the value that has been achieved by tests for ferritic and martensitic steels), the total operation period of FEA is about 30 years (that is three times over that gained for RI operation at the NS).

As it has been mentioned above, increasing the make-up fuel enrichment reduces these demands considerably. It should be also pointed out that the concepts of the core complete equipment, scheme and partial refuelings schedule, as they have been accepted in calculations at the phase of preliminary study, are not optimal. They need further multicriterion optimization by using calculational algorithms in which reactor dimensions are reflected equivalently. It enables to reduce the requirements for fuel elements operation. One such method has been presented in paper [30].

On the base of the results obtained the engineering design of SVBR-600 RI of the 625 MWe electric power has been carried out at EDO "Gidropress.", and the design of NPP using that RIs has been carried out at VNIPIET. Their aim was to choose the principal engineering and designing approaches and the preliminary assessment of their engineering and economical characteristics.

The results of calculation of engineering and economical characteristics of the two-block NPP with RI VVER-600 being compared with those of the two-block NPP with RI VVER-640 which has the highest safety characteristics for that type of RI, have demonstrated the following.

Capital costs for constructing the NPP along with the SNFR (the costs of initial fuel and coolant load calculated per 1 kW of electric power maintained have been taken into account) have been almost equal and the cost of electric energy has been 21% less than that of RI VVER-640.

The equality of specific capital costs, in spite of higher costs of FR start-up fuel load and coolant which are typical for RI SVBR-600, has been accounted by simplier scheme of this RI and reduction of constructing volumes, which is caused by the total absence of the primary circuit pipelines and valves, significantly less quantity of ancillary systems and systems ensuring safety, significantly less (4 times) volume of the SNFR under the same power generation, almost total absence of LRW (by the experience of operating the RI using lead-bismuth coolant at the NS) and, as a result of this, significant reduction of special chemical water purification systems which compensates the factors pointed out above. The less electric energy cost for SVBR-600 is accounted for the significantly more depth of fuel burning-up in the FR considered.

As a result of the investigations carried out, there have been gained the conclusions of the economical preference of the NPP using SVBR-600 RI in comparison with that using VVER-640 RI. Of course, these conclusions are preliminary because the level of studying the RI SVBR-600, that corresponds to the phase of engineering recommendations, defines the more error of engineering and economical calculations than that for VVER-640. However, these conclusions point to the expediency of further investigating the concept suggested along with the development of the NPP concept design. And the design of the NPP using VVER-1000 RI (V-392), that has the best engineering and economical characteristics, should be accepted as the comparison basis.

6. CONCLUSION

The analysis of experience of operating the RIs at the NSs and at the ground-based facilities-prototypes has demonstrated that it was only one accident that was caused by using LBC. Further work has enabled to solve the problem of LBC technology and ensure the reliable exploitation of the RIs primary circuits. The causes of other accidents and emergency situations were not concerned with the use of LBC, and similar accidents could have happened at any type of the RIs.

The developed measures on radiation safety have enabled to eliminate the personnel's irradiation by polonium-210 above the permissible limits not only under the normal operation conditions, but also if there are refueling, repair works, emergency coolant spillage.

The problem of multiple coolant "freezing-defreesing" in the RI has been solved. It enables to use this regime for providing nuclear and radiation safety during the RI transportation (for low power RIs) and long SNF storage.

The experience gained has enabled to design the number of small power RIs providing the most complete realization of the self-protection principle for the severest accidents and their deterministic elimination because of using the fast reactor, LBC and the primary circuit integral design (SVBR-75/100). On the basis of these RIs renovation of NPP's units served their lifetime can be carried out, regional NPHPs for energy shortage regions can be built, large power NPPs of modular type can be constructed, meeting IAEA requirements power complexes for electric energy producing and sea water desalinating can be built in developing and other countries, reactors for utilizing Pu and transmuting the long-lived minor actinides can be designed.

In order to maintain the NP competitive ability on exhausting the cheap uranium resources and necessity of using the FR with closed NFC and plutonium recycling (plutonium significantly deteriorates the ecological characteristics) there has been developed the concept of the FR cooled by LBC and using the depleted uranium as the fuel make-up. It enables to postpone the large-scale SNF reprocessing with plutonium recycling over 50 to 100 years.

The existing scales of metal bismuth production which have been restricted by bismuth usage will limit the rate of introducing the reactors cooled by LBC in NP. That is why after the period of operating the demonstrative reactors using LBC and corroborating their practical reliability, economical efficiency and safety and accepting the decision of their wide introduction into NP, the actions on increasing the bismuth production must be taken. Further, when the bismuth cost begins to deteriorate noticeably the NPP's economical characteristics (as the exploiting mines are being depleted), it will be expedient to start using LBC with lower bismuth content (e.g., as low as 10%), and then pure lead coolant, which is more difficult to exploit and has not been mastered yet.

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LIST OF ABBREVIATIONS

NPP - nuclear power plant;
NPHP - nuclear power heating plant;
RI - reactor installation;
NPI - nuclear power installation;
NS - nuclear submarine;
LMC - liquid metal coolant;
IPPE - Institute of Physics and Power Engineering;

LBC - lead-bismuth coolant;
NP - nuclear power;
SG - steam generator;
SHS - steam heating system;
EP - emergency protection;
CR - compensative rods;
TC - tight compartment;
WCR - water-chemical regime;
EC - emergency condenser;
LRP - leakage reinjection pump;
FR - fast reactor;
MPC - maximal permissible concentration;
FSA - fuel subassemblies;
PSA - probabalistic safety analysis;
SNF - spent nuclear fuel -
EPS - emergency protection systems;
SNPP - small nuclear power plant;
HPP - heating power plant;
MCP - main circulation pump;
SRD - scientific and research developments;
LRW - liquid radioactive waste;
RAW - radioactive waste;
NFC - nuclear fuel cycle;
LWR - light water reactor;
FEA - fuel elements assemblies;
LEU - low-enriched uranium;
CBR - core breeding ratio;
UWP - uranium waste pile;
EUU - efficiency of natural uranium energy potential utilization;
NU - natural uranium;
SNFR - spent nuclear fuel repository;
NVNPP - Novovoronezh nuclear power plant;
RDIPE - Research and Design Institute of Power Engineering;
NITI - Technological Research and Development Institute (Sosnovy Bor);
TSNII KM Prometey - Central Research Institute of Structural Materials;
OKBM - Experimental Design Bureau of Mechanical Engineering;
EDO Gidropress - Gidropress Experimental and Design Bureau (Podolsk);
SSC RF IPPE - State Scientific Center of Russian Federation Institute
of Physics and Power Engineering;
St.P MBM "Malakhit - Malakhit Marine Engineering Design Bureau (St. Petersburg);
GNIPKII Atomenergoproekt - All Russia Research and Design Institute;
VNIPIET - All Russia Research and Design Institute for Integrated
Power Engineering Technology;
VNIPI Promtechnologya - All Russia Scientific Research and Design Institute.

ADVANCED 4S (SUPER SAFE, SMALL AND SIMPLE) LMR

XA0056275

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Abstract

This paper describes a new nuclear power system which can be used for a greater variety of applications. The 4S liquid metal reactor has high inherent safety and passive safety characteristics. It is also easy to operate, maintain and inspect, faster to construct, more flexible in location, requires less initial investment, and is better suited to electrical grid management. The reactor offers a new route through which to expand the use of safe nuclear technology in the world.

1. INTRODUCTION

In the 21st century, we will be confronted with how to solve some very serious problems which have not been faced by mankind in the past. That is the so called trilemma problems energy security, environmental protection and socio-economic development which compete against each other.

Nuclear power appears to have the potential to help to solve these problems. However, to make it a reality, it is necessary to improve nuclear power technology to make it more applicable to a greater variety of utilizations and locations.

To achieve this target, the following key items are essential.

a) Enhancing the efficiency of nuclear power utilization
b) Increasing the flexibility of nuclear power siting
c) Assurance and improvement of safety and reliability
d) Improving the economics
e) Promoting public acceptance of nuclear power
f) Nuclear power utilization in developing countries
g) Improving the fuel cycle and waste management
h) Providing the high proliferation resistance

2. BASIC CONCEPT

Based on the requirements, one of the most promising nuclear reactor designs is a small or medium size modular type nuclear reactor with high inherent safety and passive characteristics. It is preferable that the following features are taken into considerations; greater simplicity, easy to maintain, inspect and operate, less influence of human factors, high reliability, improved availability and capacity, design standardization, easier to construct, quicker to construct, more flexibility in siting, lower initial investment and better adaptability to electrical grid management.

All these features of nuclear power plants make it very attractive for developing countries to introduce nuclear power and for industrialized countries to expand nuclear power usage. In addition, the need to improve the fuel cycle, waste management and nuclear proliferation resistance, requires special design characteristics and fuel management provisions.

Our design efforts to satisfy these conditions have resulted in the development of the Super Safe, Small and Simple (4S) fast reactor with an electric power output of 50 MWe.[1),2),3),4),5)] There are several advantages of the 4S reactor:

a) The 4S will play an important role in accelerating nuclear power utilization throughout the world, because the reactor provides an effective power supply on isolated locations, and medium or large power stations are also feasible by the core or reactor module configuration.

b) Regarding site selection, the 4S has some favorable characteristics. First, the required site boundary is 20m based on the source term evaluation and thus the 4S plant satisfies the required area by it's construction area. Second, the small reactor vessel has a high seismic resistance. The 4S can be constructed to withstand any earthquake condition without changing the main design. Third, as the reactor is embedded, external accidents such as falling aircraft do not pose serious safety problems.

c) Higher safety can be achieved by designing all reactivity feedback coefficients including coolant void reactivity to be negative and by controlling neutron leakage from the core by an annular reflector. The potential for super prompt criticality, particularly during start up, is completely excluded by using metallic fuel. A fully passive heat removal system is employed in the 4S so that the auxiliary support system for the safety system can be eliminated, thus improving the reliability of the safety system.

d) The 4S is made more economical by simplifying the entire plant design. In the reactor assembly, control rod drive mechanisms, a rotating plug for refueling and a refueling machine are not necessary. All auxiliary systems in the nuclear building are eliminated by removing heat by natural air circulation. No complex control system is required. These simplicities greatly reduce the cost of the 4S plant.

e) In order to allay public fears, "a sense of security" is essential, which means that a clearly safe concept, proven or easily demonstrable technology and small system technology are preferable. The safety of the 4S can easily be demonstrated in a full scale test because of its small size. As the 4S is a modular type reactor, the full advantages of modular reactors are directly applicable. We believe that a simple system like the 4S will be readily understood and accepted by the public.

f) In order to introduce nuclear power plants to developing countries, the plant should be easy to operate and require less maintenance. These are both features of the 4S plant. Completely automatic operation is possible because any malfunctions during start up and power generating operation do not cause severe reactivity insertion. There are no rotating parts which require frequent maintenance. The coolant can be driven by electromagnetic pumps. Because the safety systems are passive, no active tests are required. These factors greatly reduce the number of operators required. In addition, the refueling interval is ten years, and this also reduces the fuel exchange workload on operators. Other requirements for reactors in developing countries are related to environmental problems. Reactors need to address the problems faced by such countries in the future, which include rapid population growth, carbon dioxide production and desertification.

g) Improving the fuel cycle and waste management is most important for future nuclear systems. A fast reactor technology using a metallic fuel cycle appears to be a most promising approach.[6] The technology is valuable because it has the potential to simplify reprocessing, fuel fabrication process and nuclear waste disposal, and it includes actinide recycling which is important from the viewpoint of resource utilization. It also reduces the fuel cycle cost dramatically.

h) In order to keep strict control over the plutonium used, the 4S incorporates a new concept by using metallic fuel which significantly helps to achieve the non-proliferation goal; a large amount of fuel can be confined for a long time in the reactor vessel without refueling. During the initial start up, the reactor is sealed in the presence of IAEA (International Atomic Energy Agency) authorities and the IAEA maintain long-term control over operations to ensure non-proliferation.

The 4S reactor assembly and plant design are shown in Fig. 1 and Fig. 2, respectively. The diameter of the reactor vessel is 2.5 m and the area of the nuclear building is 26m x 16m, thus requiring only a small ground space.

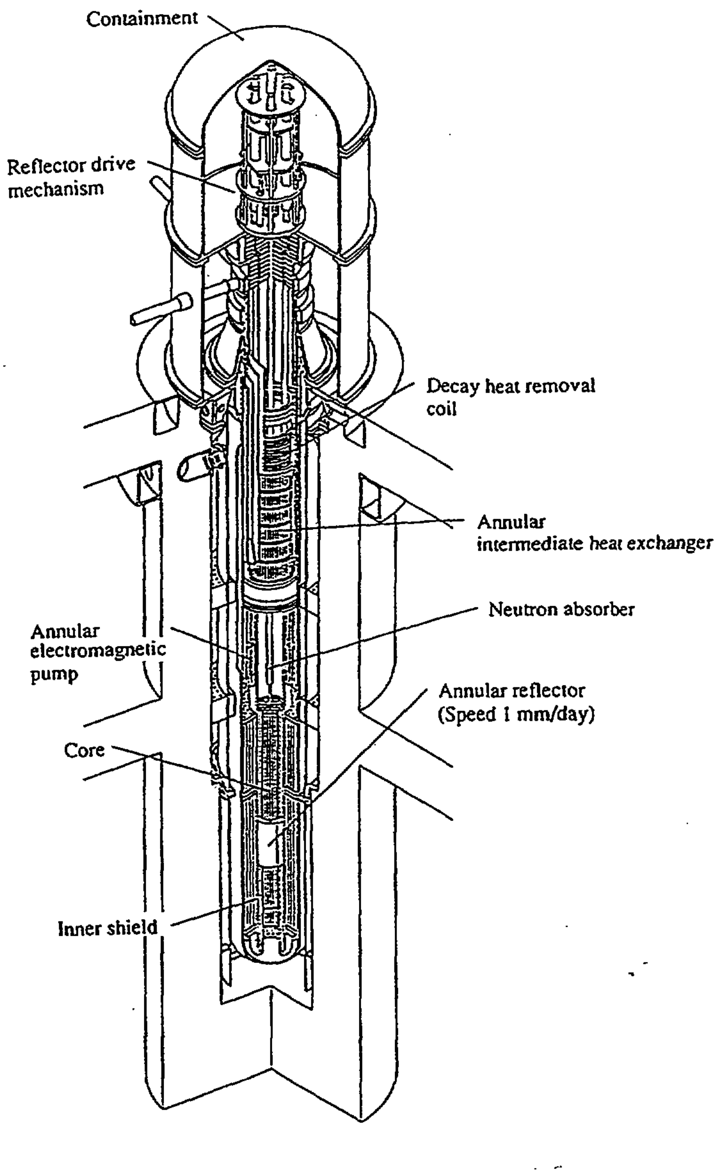


Fig. 1 4S Reactor Assembly

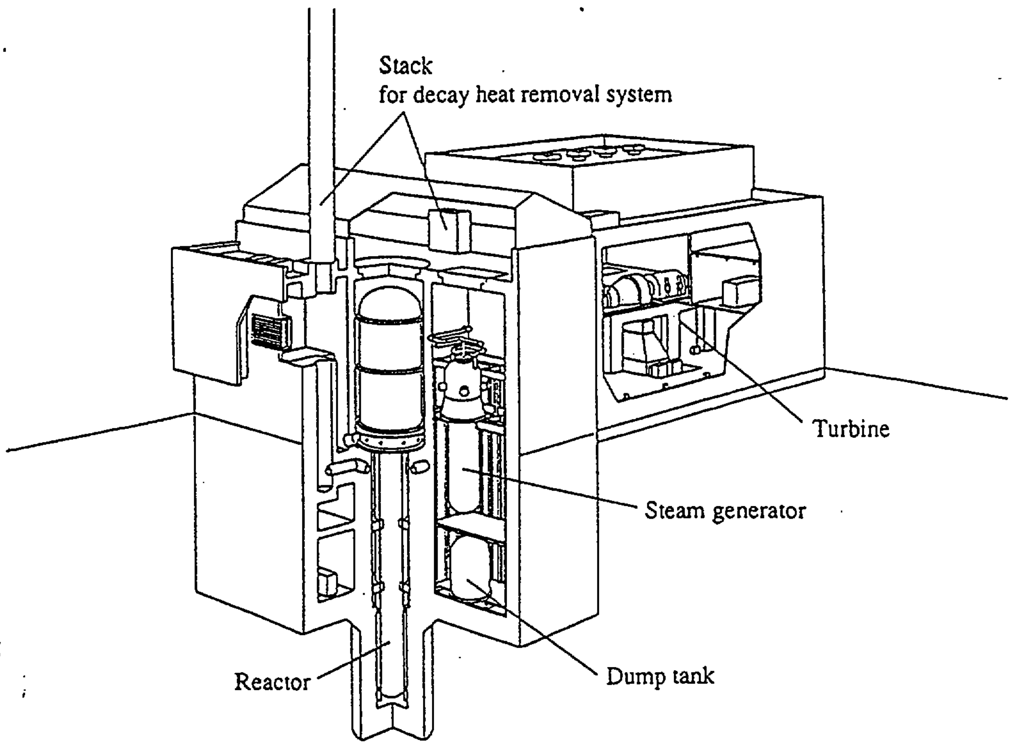


Fig. 2 4S Plant Concept

3. CORE AND REACTOR SYSTEM

3.1. Core and Reactivity Control System

The 4S employs a reactivity control system with an annular reflector in place of the control rods and driving mechanisms which traditionally require frequent maintenance service. Reactivity is controlled only by the vertical movement of the annular reflector during plant startup, shutdown and power generation, thus eliminating the necessity for complicated control rod operations. Although this reactivity control method using a reflector has been studied in some projects[7),8)], using this method for the core burn-up phase is a new approach.

The reflector is installed inside the reactor vessel and the heat generated in the reflector is cooled by sodium. The equivalent core diameter is 0.8m which satisfy negative void reactivity requirements. The reflector length is 1.5m and the reflector gradually moves up to control the reactivity leading to burn-up. The axial power distribution changes as shown in Fig. 3 according to the reflector position.

The structure of the reflector is shown in Fig. 4. The upper part of reflector must be made of a material with a lower reflection effect than the coolant itself in order to increase its ability to control neutron leakage. The reflector therefore has a gas cavity which can increase 3% Δk of the reflector reactivity compared to a reflector without a cavity.

Table 1 lists the major specifications of the core and fuel. The cross section is shown in Fig. 5. The core has an active length of 4m.

The fuel inventory balance is shown in Table 2. The amount of $Pu^{fissile}$ is 1.3tons at start up of the 4S core and 1.2tons remain after ten years of operation. In this sense, the 4S core could be called a plutonium burning storage core.

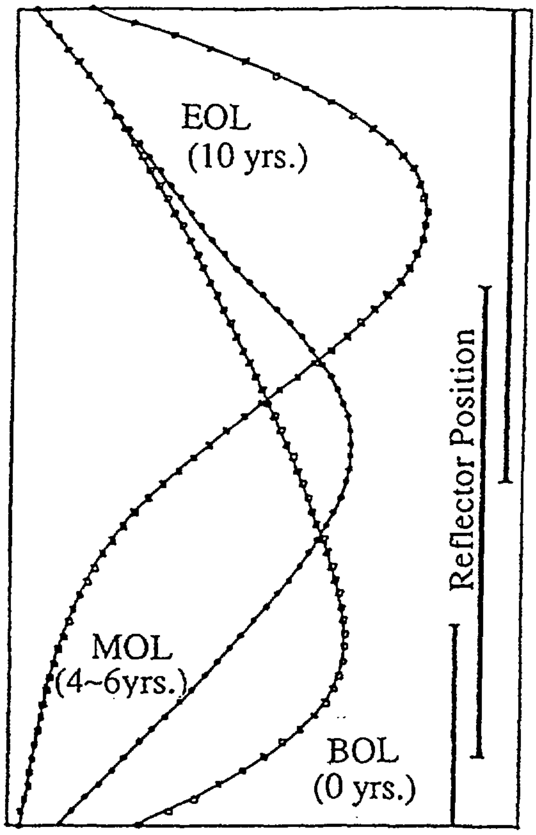


Fig. 3 Axial Power Distribution as a Function of Reflector Position

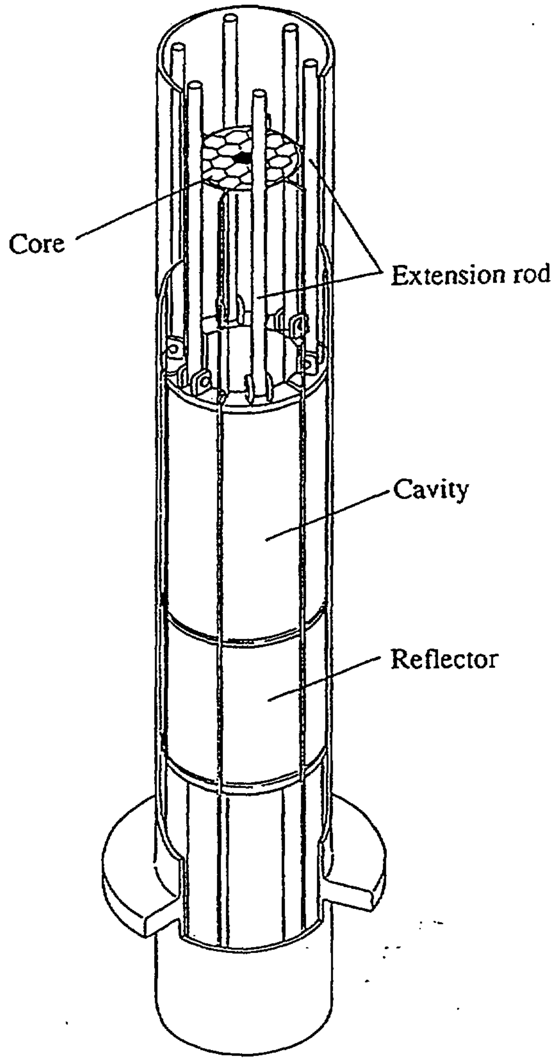


Fig. 4 Structure of Reflector

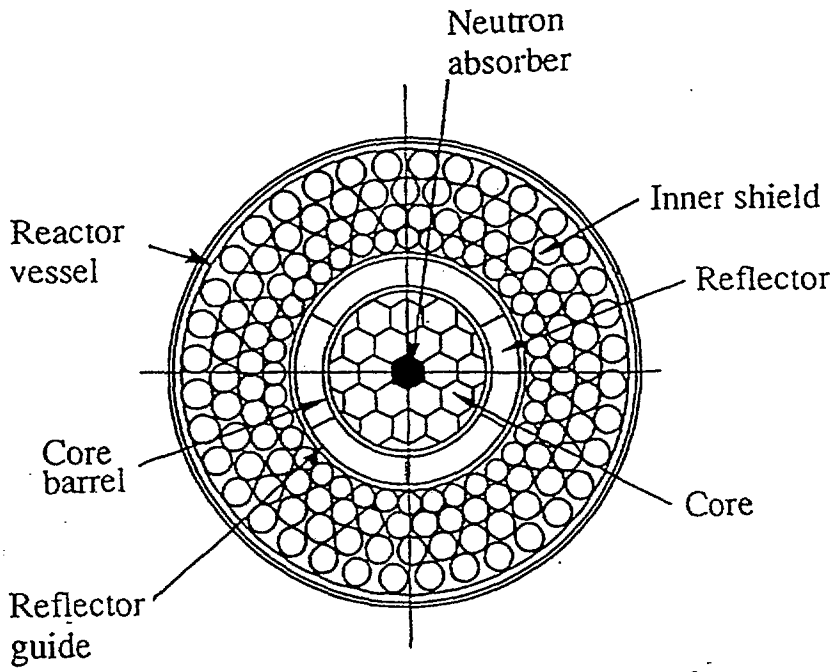


Fig. 5 Cross Section of Core and Reactor Vessel

Table 1 Major Core Design Parameters

FUEL COMPOSITION (Metallic Fuel)	U·Pu·Zr or U·Zr
CONVERSION RATIO	0.7
VOID REACTIVITY	–2.5$ (Diffusion Model) –1.0$ (Transport Model)
REFUELING INTERVAL	10 years
Pu^{fiss} INVENTORY	1.3 ton
No. of SUBASSEMBLIES	18
No. of PINS/S/A	217
PIN DIAMETER	1 cm
PIN LENGTH	610 cm

Table 2 Fuel Inventory Balance

	BOL (kg)	EOL (kg)	BOL-EOL (kg)	FISSION CAPTURE DECAY (kg)	FISSION (kg)	CONTRIBUTION TO TOTAL FISSION (%)	CAPTURE (kg)	DECAY (kg)	REMARKS
U^{235}	23.9	17.4	6.5	6.5	5.2	1.1	1.3		neutron capture
U^{236}	0	1.3	–1.3						
U^{238}	7946.0	7621.2	324.8	324.8	65.0	14.3	259.8 (a)		neutron capture
U^{TOTAL}	7969.9	7639.9	330.0	331.3	70.2	(15.4)			
Pu^{239}	1272.9	1143.6	129.3	389.1 (*1)	326.8	71.8	62.3		neutron capture
Pu^{240}	520.7	510.6	10.1	72.4	38.9	8.6	33.5 (b)		neutron capture
Pu^{241}	38.9	37.8	1.1	34.6	14.0	3.1	2.3	18.3	neutron capture & decay
Pu^{242}	96.4	88.7	7.7	10.0	5.0	1.1	5.0		neutron capture
$Pu^{fissile}_{Total}$	1311.8	1181.4	130.4	423.7 (c)	340.8	(74.9)			
Pu^{Total}	1928.9	1780.7	148.2	506.1	384.7	(84.6)			
Total Heavy Metal	9898.8	9420.6	478.2		454.9	100			

(*1) 129.3+259.8 Breding Ratio= $\frac{(a)+(b)}{(c)}$ =0.69

3.2. Reactor Assembly

The major specifications of the reactor are shown in Table 3. The primary coolant flow path is shown in Fig. 6. The coolant flows out of the core, rises in the hot pool and

Table 3 Major Reactor Design Parameters

THERMAL POWER	125 MWth
REACTOR OUTLET TEMP. INLET TEMP.	510°C 355°C
REACTIVITY CONTROL	ANNULAR REFLECTOR
PRINARY PUMP	TWO ANNULAR SINGLE STATOR EMPs JOINED IN SERIES
INTERMEDIATE HEAT EXCHANGER	ANNULAR STRAIGHT TUBE TYPE
VESSEL DIAMETER	2.5 m
VESSEL THICKNESS	25 mm
VESSEL MATERIAL	SUS 304
CORE INTERNAL MATERIAL	Mod9Cr-1Mo

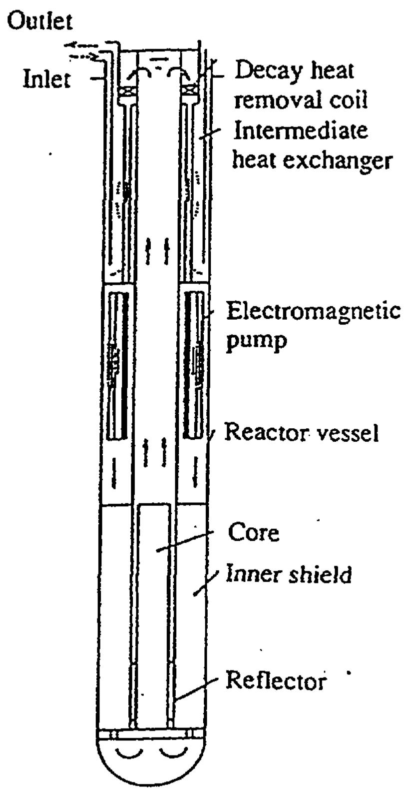


Fig. 6 Primary Coolant Flow Path

descends in the intermediate heat exchanger through which the heat is transferred to the secondary system. It is pressurized by a primary electromagnetic pump at the bottom of the intermediate heat exchanger and flows down in the annular space. Then, the coolant turns up at the bottom of the reactor vessel and enters the core.

The intermediate heat exchanger (IHX) and the electromagnetic pump (EMP) have an annular shape and an annular short vertical redan is installed to form an annular flow path. A space is provided outside the core barrel, in which the reflector moves vertically. An upward flow path for the coolant is formed in this space to remove the heat generated in the reflector body.

The primary pump is composed of two annular single stator EMPs joined in series. The feasibility of this type of sodium-immersed, self-cooled EMP has already been established by a small model.[9)]

The reflector driving mechanism consists of a hydraulic system which operates at start up and shutdown and a ball screw that is connected to a motor which is operating during normal operation. The mechanism has six driving systems corresponding to the number of reflector segments.

The reflector is moved upward by the hydraulic pump during start up. During power operation, the reflector is held by the hydraulic system and gradually moves up for burn-up compensation at a constant speed of 1mm/day without any speed control system. To attain this very slow speed, a reduction mechanism composed of paradox planetary gears is installed. The technical reliability of the gears has been demonstrated elsewhere. However, a spare set of gears is installed in the 4S in case of trouble. To shut down the reflector, the scram valve is opened in the hydraulic circuit. When the reflector lowers 1m, the core reaches the subcritical cold shutdown state. The length of the downward movement of the reflector is determined by the capacity of the hydraulic cylinder. It cannot move otherwise.

A natural air cooling system is employed to cool the reactor cavity. While its main purpose is to remove heat from the reactor during normal operation, it also functions to remove decay heat.

4. INHERENT AND PASSIVE SAFETY

4.1. Inherent core safety

4.1.1. Negative void reactivity

One of the attractive features of the fast reactor is its hard neutron spectrum. To expand this feature, a metallic fuel core is employed in the 4S. However, it is more difficult to reduce void reactivity for a core with a harder spectrum. It is very important to design the void reactivity to be negative in order to prevent a severe nuclear accident in the event of sudden loss of coolant, sudden loss of coolant flow or a large gas bubble entrainment in the core.

There are two generic approaches to reducing void reactivity. Reducing the core height is one popular approach, and a core with a small diameter is another effective method. In the 4S, making the core diameter small is the preferred approach because this reduces the vessel diameter and enhances the value of the reflector reactivity. By reducing the core diameter, neutron leakage is enhanced in the radial direction so that negative void reactivity is maintained during the entire core life time. The void reactivity depends on the diameter and fuel volume fraction as shown in Fig. 7.

For the selected core, the void reactivity of the total core is -1$ at the end of life based on the transport calculation. Other temperature feedback coefficients are all negative as shown in Table 4.

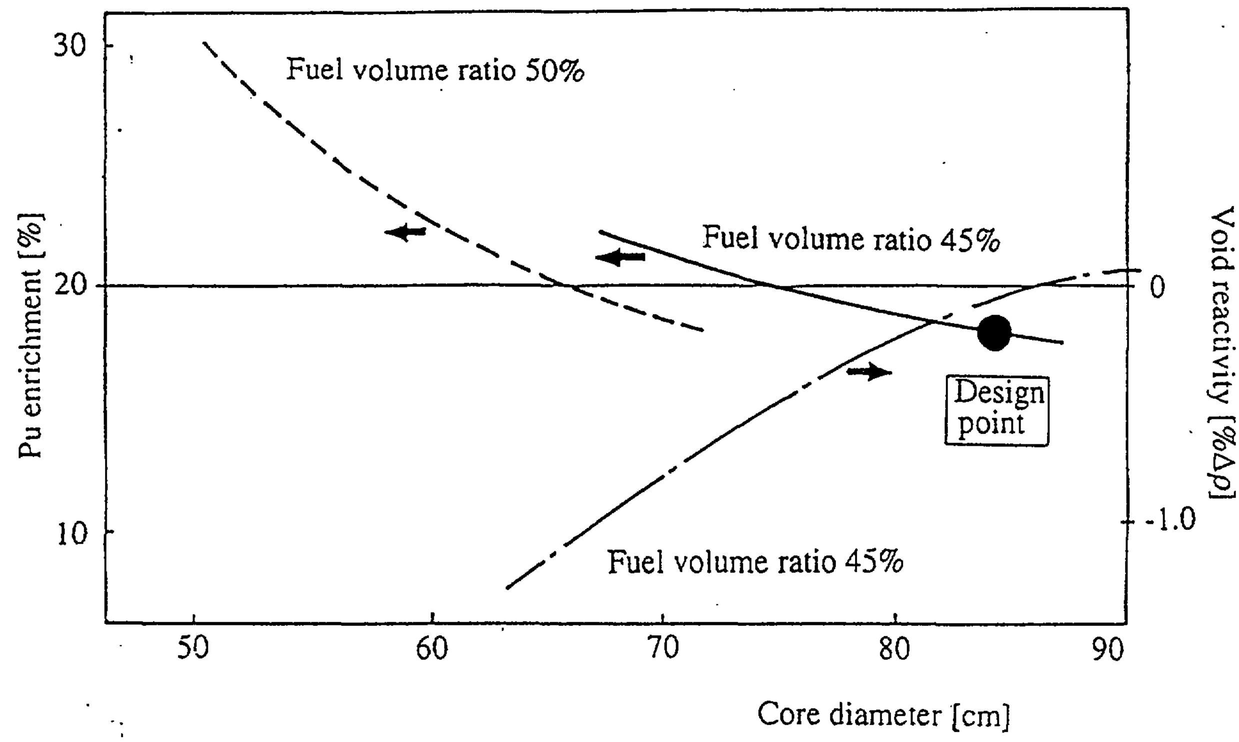


Fig. 7 Design Point for Ten Year Core with Negative Void Reactivity

Table 4 Feedback Temperature Coefficients

	BOL (BARE) (0 YRS.)	BOL (REFLECTOR) (0 YRS.)	MOL (4~6 yrs.)	EOL (10 yrs.)
FUEL $\left(\frac{K/KK'}{°C}\right)$	-8.87×10^{-6}	-8.23×10^{-6}	-7.37×10^{-6}	-7.29×10^{-6}
STRUCTURE $\left(\frac{K/KK'}{°C}\right)$	-1.62×10^{-6}	-1.30×10^{-6}	-0.42×10^{-6}	-0.50×10^{-6}
COOLANT $\left(\frac{K/KK'}{°C}\right)$	-6.03×10^{-6}	-5.22×10^{-6}	-2.87×10^{-6}	-3.23×10^{-6}
CORE SUPPORT $\left(\frac{K/KK'}{°C}\right)$	-8.34×10^{-6}	-7.87×10^{-6}	-6.84×10^{-6}	-6.70×10^{-6}
DOPPLER $\left(T\frac{dk}{dT}\right)$	-1.83×10^{-3}	-2.22×10^{-3}	-2.79×10^{-3}	-2.80×10^{-3}

4.1.2. Preventing potential super prompt criticality.

It is essential for the safety of the reactor to exclude the possibility of super prompt critical state at all times. This requires that the inserted reactivity at potential events should be below 1$ under conservative conditions, neglecting reactivity feedback coefficients.

The largest reactivity change occurs during plant start up. The reactivity decrease from criticality at zero power under cold temperature conditions to full power is generally above 1$. The worst case is reactivity insertion under cold temperature conditions.

At plant start up in the 4S, the system temperature is raised to 350°C by heat input from the electromagnetic pump before raising the reflector. This procedure greatly reduces the reactivity temperature swing. The reactivity to be inserted to increase the power is about 86¢, which causes the following reactivity effects; thermal expansion of the fuel, structure, coolant, core support grid and doppler reactivity. Because metallic fuel is employed in the 4S, the reactivity is small compared with the 150¢ for MOX (Mixed Oxide) fuel, mainly due to its small Doppler coefficient.

The basic dynamic characteristics of the core under various reactivity insertion conditions are shown in Fig. 8. The power transient reflects the super prompt critical condition when a large reactivity insertion occurs. On the other hand, the power transient is small for the 4S during potential reactivity insertion at the plant start up phase.

4.1.3. Neutron leakage control by reflector.

In the 4S core, all reactivity change is controlled by the reflector. This neutron leakage control system has a decisive advantage compared with control rod system from the safety point of view.

The active length of the core is 4m, which is surrounded by a 1.5m long reflector. The reflector is separated into six azimuthal parts, each of which can move from the bottom to the top of the core along with the core burn-up. If an uncontrolled lift of each part of the reflector occurs, the core criticality cannot be sustained. The new geometry of the reflected region causes negative reactivity insertion because of the enhanced neutron leakage.

Figure 9 shows that lifting up parts of the reflector gives strong negative reactivity except lifting up all of the reflector. Although the figure shows negative insertion, a small positive insertion up to ten cents may be possible if each segment of the reflector moves up a small distance from the original position. The maximum is -4$ when three parts are lifted up. This geometry gives the minimum criticality which maximizes neutron leakage.

Thus, the inherent core safety against partial movement of the reactivity control system is assured for the 4S core.

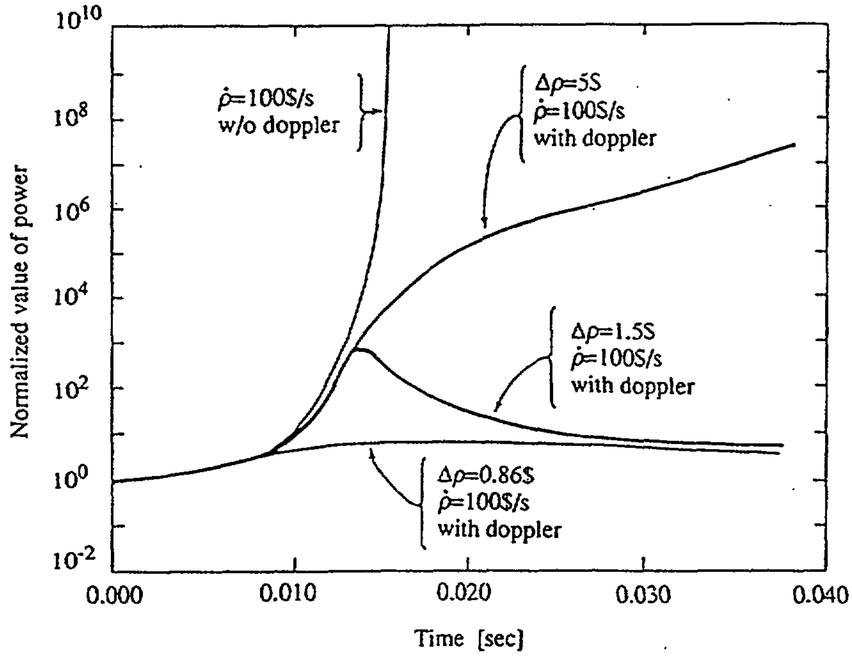


Fig. 8 Reactor Power Transients for Various Reactivity Insertion with ρ =100$/s

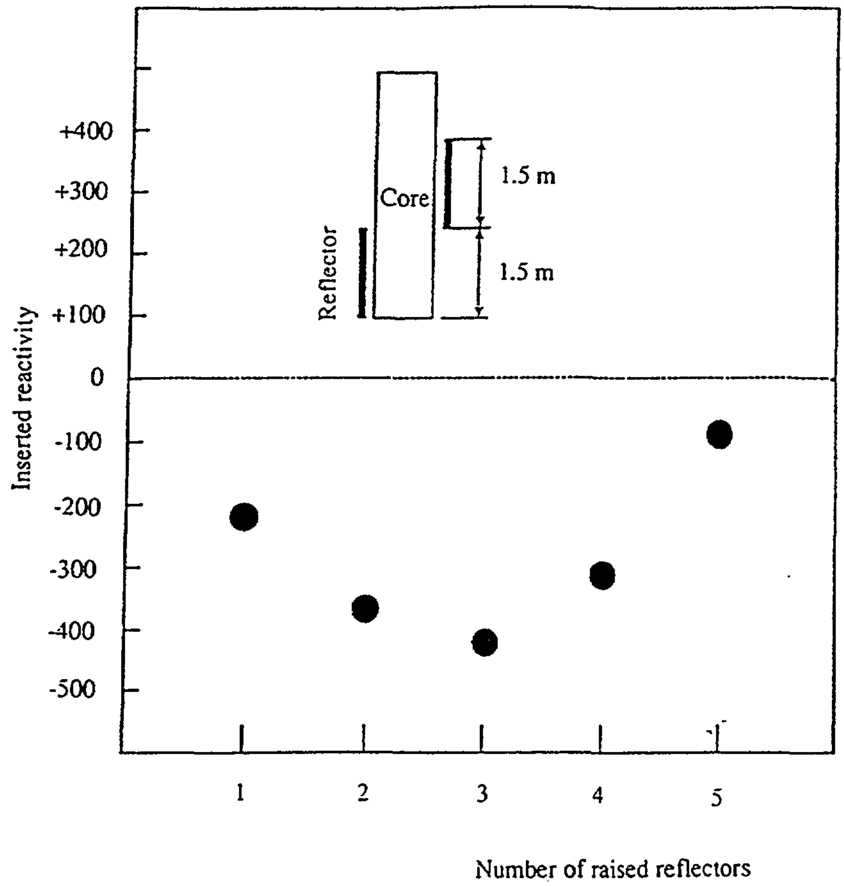


Fig. 9 Reactivity Insertion when Lifting Partial Segments of Reflector up to 1.5m

4.2. Passive Safety for Unlikely Events

4.2.1. Loss of all station power without scram.

In the 4S, following the loss of off-site power, the primary loop shifts to natural circulation. The pump in the secondary loop of the decay heat removal system does not work assuming the loss of emergency power following loss of off-site power. Under such conditions, the secondary coolant operates in natural circulation mode and natural air-cooling operates in the air cooler. Thus, a passive heat removal circuit is established.

In the safety analysis, all active shutdown systems are assumed to fail, and negative feedback coefficients are taken into consideration for the analysis. Figure 10 shows the analysis results. The primary natural circulation flow rate is about 10% of the rated flow due to the large distance between the core and decay heat removal coil. The reactor power is decreased by the negative feedback coefficients and the transient peak temperature of the cladding is 810°C, dropping to 510°C after 150 seconds. No cladding damage is expected during the accident.

4.2.2. Loss of decay heat removal function.

In the 4S, the decay heat is removed by two systems consisting of the decay heat removal coil installed in the reactor (PRACS) and the natural air ventilation from outside the guard vessel (RVACS). The analysis considers the destruction of PRACS and the RVACS cooling stack by a large falling aircraft. In addition to this extreme severe condition, 50% of the cross sectional area of the RVACS stack is assumed to be blocked.

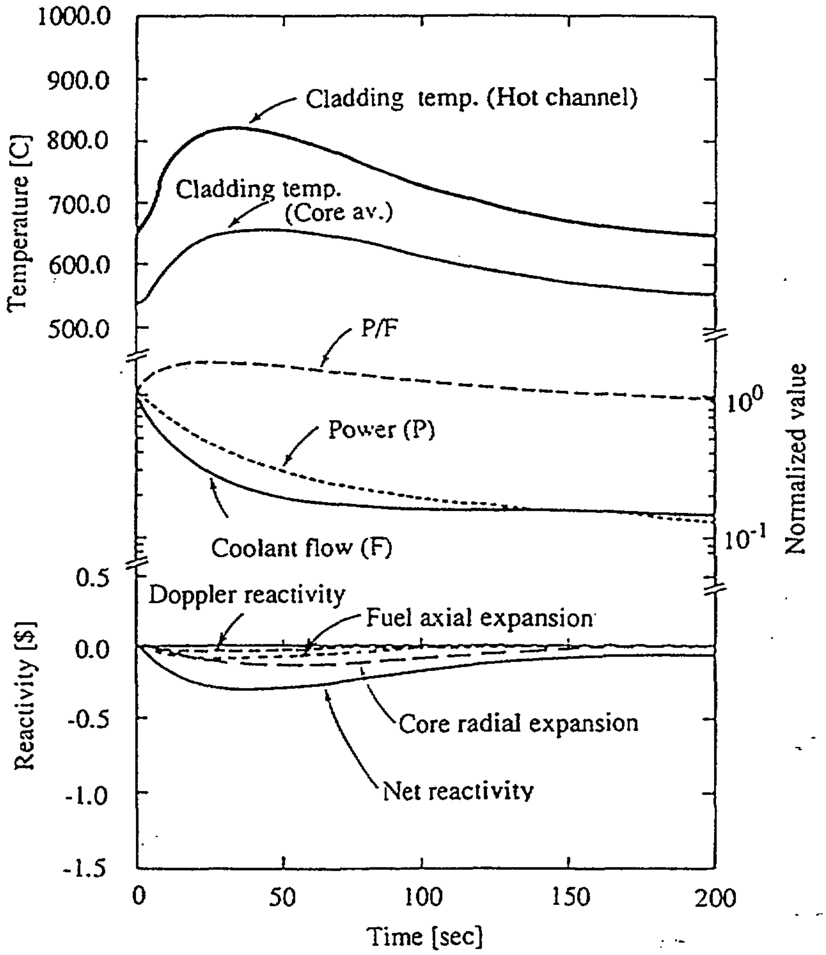


Fig. 10 Core Response for Loss of all Station Power without Scram

Figure 11 shows the analysis results of the primary coolant temperature. The coolant temperature gradually rises and peaks at 700°C after 35hrs. The structural integrity is a major concern in this type of accident. However, a creep damage during the accident is less than 0.1 as the usage factor, and so the structural integrity is maintained.

4.2.3. Sodium-water reaction in steam generator.

The breakage of all heat transfer tubes in the steam generator was assumed, and protective actions like water steam dumping were neglected to find the passive safety features against this type of accident.

The water leak rate increases as the tubes break, but is limited by the feedwater pump capacity. After the water leak rate reaches the peak value, the main steam pressure decreases because the turbine trips due to the reduction of the main steam flow rate, and the feed water pump trips due to the reduction of its rotational frequency. Due to these quasi-passive features, the water leak rate decreases and adverse consequences can be avoided by providing two pressure relief pipings with rupture disks at the bottom of the steam generator which operate at a pressure of 11 kg/cm^2.

Figure 12 shows the pressure transients. The three lines show (1) water, steam side pressure, (2) pressure without relief piping and (3) pressure in IHX with two relief pipings. The pressure of IHX in line (3) is found to be within the allowable pressure of 9 kg/cm^2.

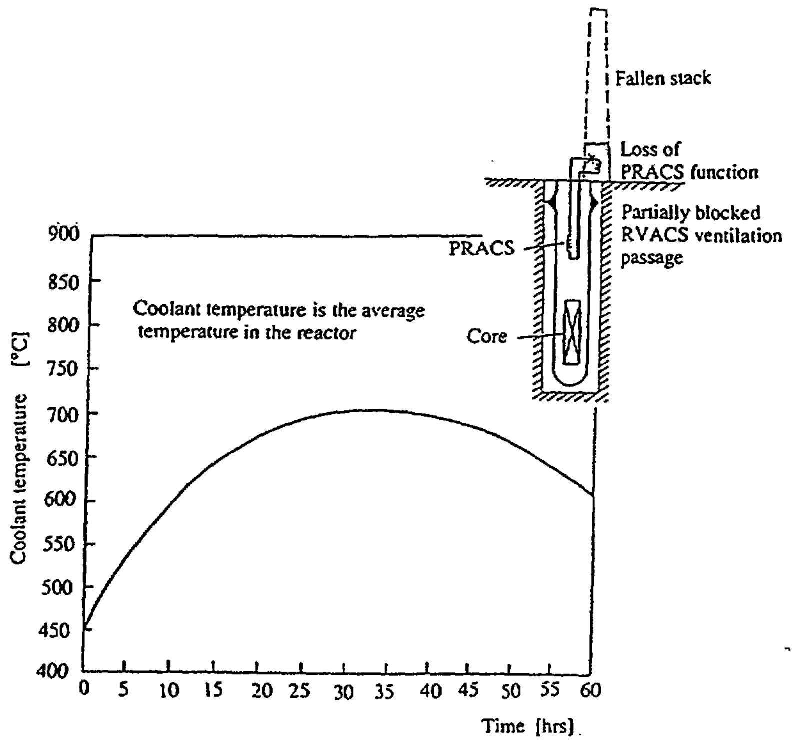


Fig. 11 Primary Coolant Temperature Response upon Loss of Decay Heat Removal System

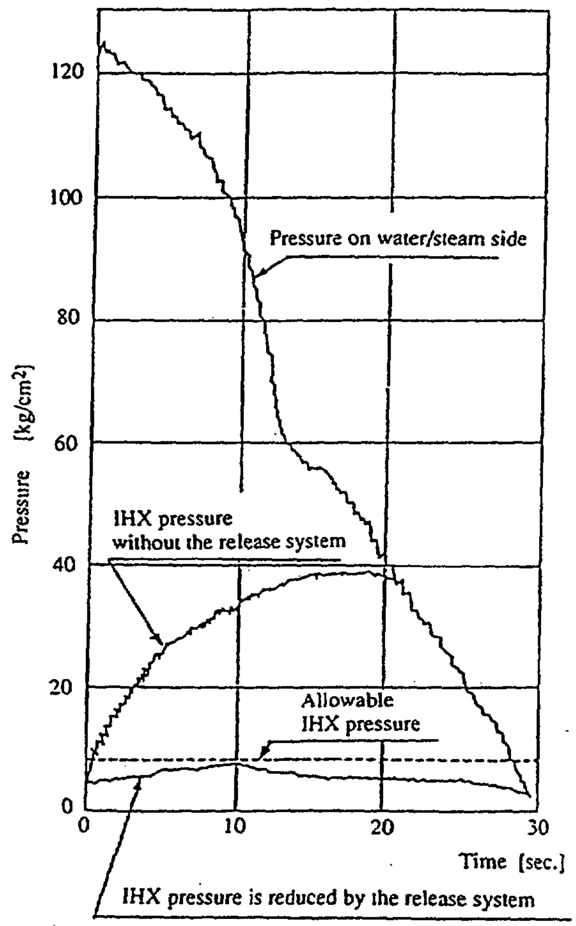


Fig. 12 Pressure Transients upon Total Tube Breaks in Steam Generator

5. OPERATION

One of the excellent features of the 4S is that it is simple to operate. There are no feedback control systems and no human intervention is required. All reactivity control is performed by the automatic movement of the reflector as shown in Fig. 13.

The plant starts up by heat entering from the primary pump and the system temperature rises to 350°C from the cold shutdown state. Under this condition, all parts of the system, including the recirculation line in the water system, are uniformly heated. Then, a neutron absorber at the center of the core is withdrawn. At temperatures below 350°C, the neutron absorber cannot be withdrawn by the self-connected mechanism using the thermal expansion difference between the stainless steel and Cr-Mo steel (Fig. 14). After withdrawal of the neutron absorber, the reflector is lifted up by the hydraulic system to reach critical condition at 350°C. A fuzzy control system is employed for this approach and a fully automatic operation circuit is provided because no malfunction causes severe reactivity insertion as described previously.

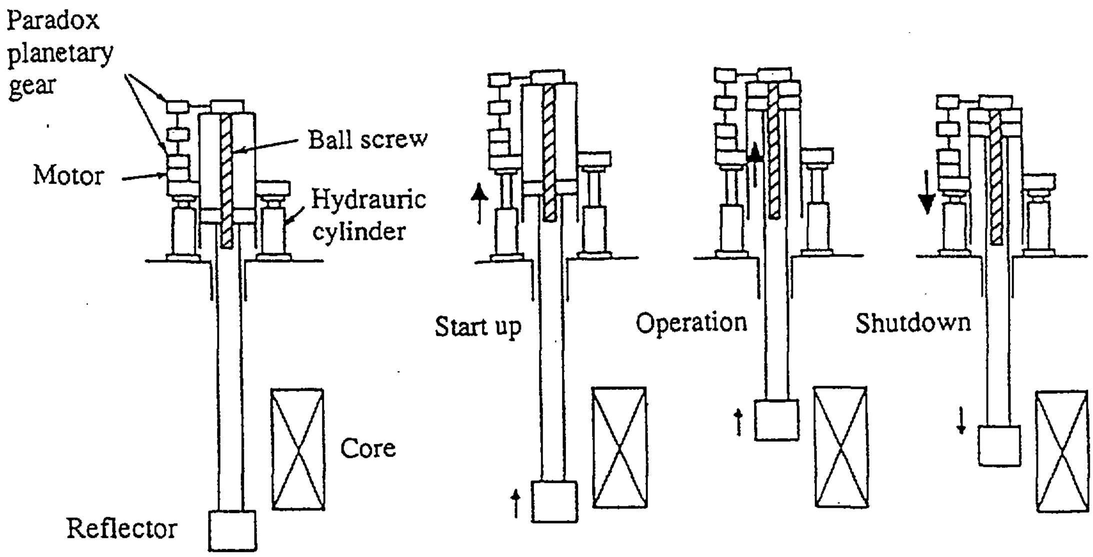


Fig. 13 Reflector Drive Mechanism and Reflector Position During Plant Operation

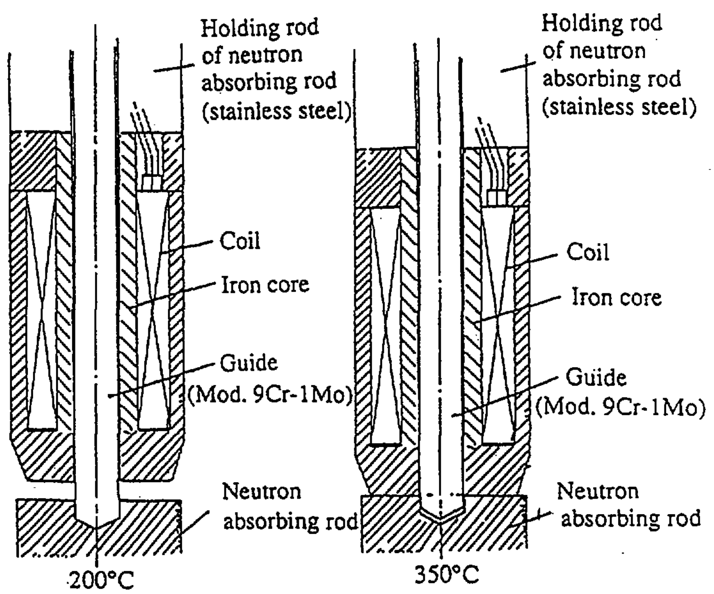


Fig. 14 Self-connected Mechanism

To increase the power to 20% of the rated power to start the turbine generator system, the reflector is periodically lifted up at a speed of 1mm per 15 minutes in automatic mode. Periodic operation is needed to stabilize the system heat balance. At the same time, the pressure of the water steam separation tank is decreased to generate steam and to reduce the re-circulation flow. After this, the power is increased to full power by lifting up the reflectors and increasing the feed water flow.

Regular power operation is attained by moving the reflector upward at a constant speed of 1mm/day to compensate for the reactivity decrease due to the burn-up of the core. Since no feedback system or control system are used, the reflector speed remains constant and the electric output is adjusted by varying the feed water flow rate to control the core inlet temperature. The controllable range of the power level by the water flow is ±10% at the rated power, which is limited by the steam generator heat balance. Beyond this range, a back-up control mechanism to adjust the reflector position is installed in the driving mechanism.

To follow the load, the core inlet temperature is changed by controlling the water flow so that the generator output coincides with the load-following control, thus causing the reactor output to follow.

Figure 15 shows the changes in system parameters as a function of time in the event of a sudden 20% loss of power. It takes 10 minutes for the reactor to shift to the new plant conditions corresponding to the load since each system or component has a time lag due to thermal inertia. No movement of the reflector is required.

As mentioned above, elimination of all feedback control systems from the reactor and secondary heat transport systems makes the 4S plant control system very simple and economic.

6. MULTIPLE USES OF THE 4S

6.1. Sea Water Desalination

A design study has investigated using the 4S to supply drinking water and other service water in regions where serious water shortages are forecast to occur in the next century.[10] The study has been extended to create green belts in desert areas in order to stop further desertification and to create more greenery to absorb CO_2.

6.1.1. Use to supply drinking water.

The concept of a nuclear seawater desalination plant is shown in Fig.16. The sea water desalination plant is planned based on a two stage reverse osmosis system with a capacity of 240000m^3/day x 7 lines by using a single 4S plant. The plant can be constructed on a site of about 210m x 140m.

Since the island system is employed, an auxiliary power facility of 5000kW is required to startup the 4S and a gas turbine plant is constructed for this purpose. The electric power consumption of the reverse osmosis system is about 5kWh/m^3. Assuming 1kWh/m^3 is required to pump the water produced, about 6 kWh/m^3 in total is required to operate the sea water desalination plant. Since the house load of the nuclear plant is 4100kW, a total power of 46MWe is required for the plant.

The nuclear power plant used is the 4S plant shown in Fig. 2. Refueling is not required for ten years, thus greatly reducing the refueling workload. The plant operation of the 4S, including full automatic start up without any control system, eliminates safety problems. These are attractive features, particularly for developing countries.

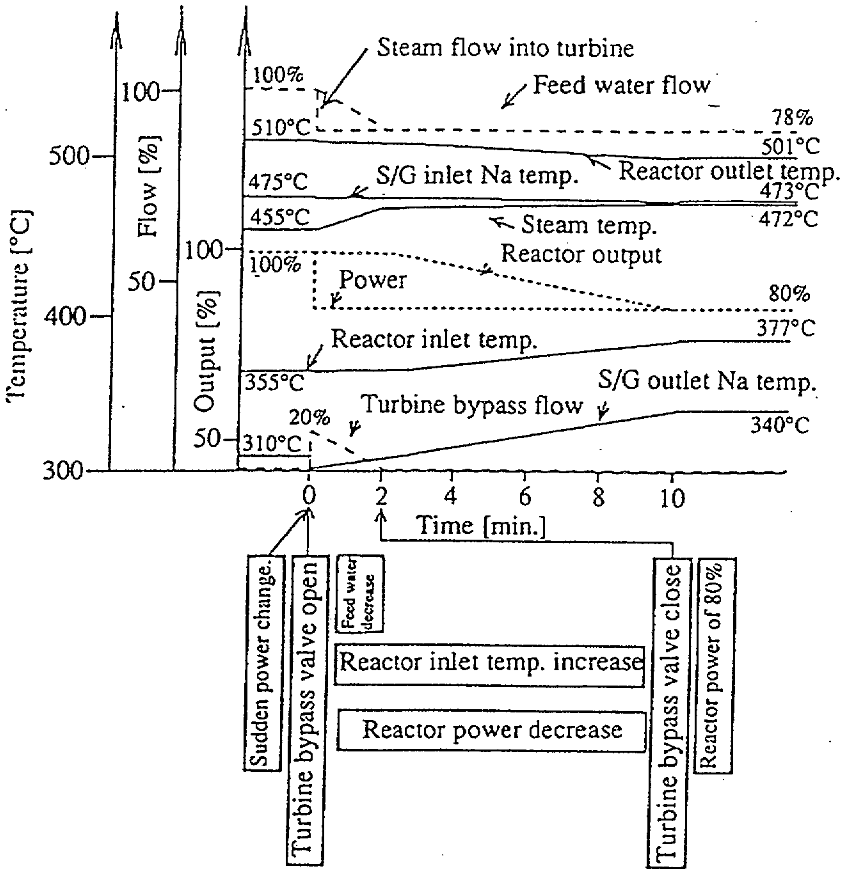


Fig. 15 System Parameters as a Function of Time for a Sudden 20% Loss of Power

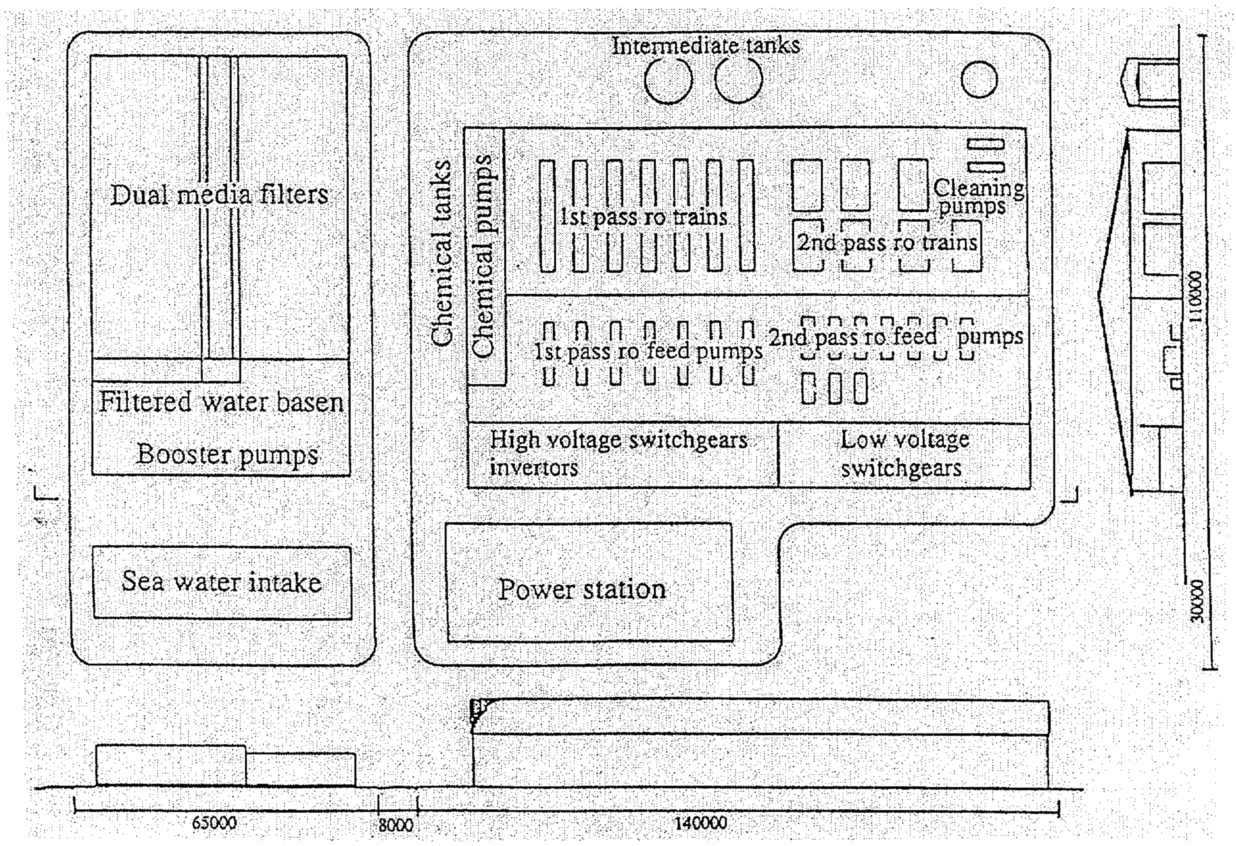


Fig. 16 Plan View of Nuclear Desalination Plant with Water Production Capacity of 24000 $m^3/d \times 7$ Lines

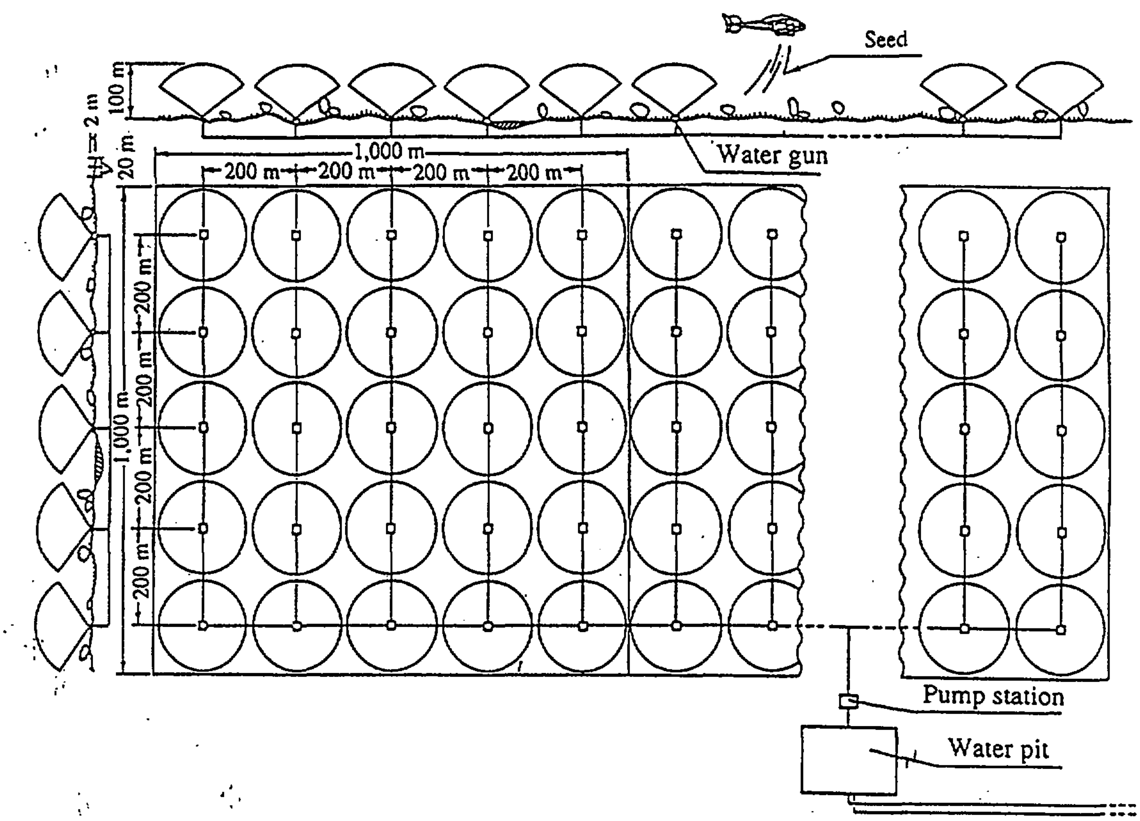


Fig. 17 Schematic of Water Supply Units

6.1.2. Creating green belts in desertification areas.

Six million hectares are devastated annually in the world by a process known as "desertification". To control desertification, we need to create grasslands or green belts at the leading edge of the desert area. A dual-purpose plant for electric power and water production is significant in this case because the desalination power plant can also supply surplus power to the desert areas. This helps to preserve forests when energy requirements in such areas are met by wood fuel, preventing desertification in a double sense.[1)]

The sea water is desalinated by two 4S units with a combined power of 100MWe, of which 10MWe is supplied to neighbouring towns for electricity and 10MWe for power for spraying water. If the remaining 80MWe are used for desalination, water can be desalinated at a rate of about 30 x $10^4 m^3$/day.

Assuming that 1000-2000mm/year of sprayed water is required to grasslands, taking evaporation in the desert into consideration, a green belt 1km wide and 100km long can be created. By locating 100m class spray guns at intervals of 200m, the plan requires 2500 guns to be installed to create a green belt 100km long and 1km wide shown in Fig. 17.

6.1.3. Renewal of the earth.

According to documents from the Intergovernmental Panel on Climatic Change, the greenhouse effect is likely to have a far more serious impact than expected. In the worst case, the carbon dioxide produced by developing countries, whose total population is expected to reach seven billion by the year 2030, could exceed 13 billion tons per year.

If it were possible to transform approximately 28 million km^2 of desertified area to at least grassland condition, this could create a carbon dioxide absorbing capacity of 14 billion tons/year, since grassland fixes absorbed carbon dioxide in the form of leaves, and stems at the rate of $500g/m^2$. Thus, the vegetation recreated in desertified areas would be sufficient to absorb and fix the carbon dioxide produced in 2030. However, the power required to create such an immense area of grassland by sea water or underground water desalination is too large and is not realistic. A realistic approach may be 1) suppression of CO_2 release, 2) balanced population growth, and 3) full use of dual-purpose nuclear power plants for electric power and water production. The nuclear plant, in this case, should be safe, and easy to operate like the 4S.

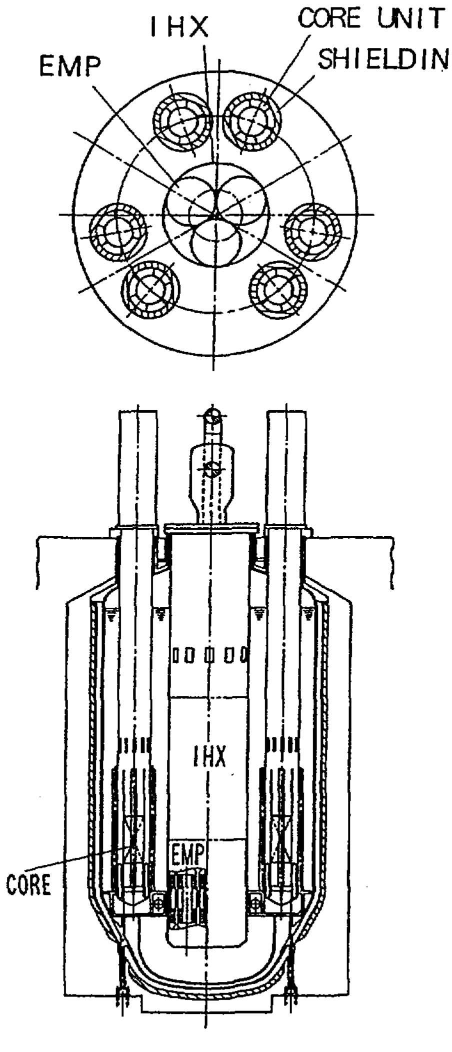
6.2. Integral concept

To minimize the time and cost for development, a modular core assembly system is preferable so that the reactor electric output can be expanded only by adding 50MWe core units in the reactor vessel. This requires the development of a single 50MWe core unit. Once developed, the reactor output can be increased without further R&D effort.

6.2.1. Modular Core Assembly Concept.

By arranging six nuclear de-coupled core units within the reactor vessel, the economy of scale increases while maintaining the safety of the 4S. The electric output is 300MWe, with six 50MWe cores to eliminate the need for refueling for ten years.

The longitudinal cross section of the reactor is shown in Fig. 18. The diameter of the reactor vessel is approximately 9m using three electromagnetic pumps combined with an intermediate heat exchanger.



Reactor Concept of Multi Core Units(300MWe)

Fig. 18

6.2.2. Modular Reactor Assembly Concept.

Figure 19 shows the nuclear power plant with fuel cycle facility. The total electrical output is 1000MWe with 10 reactor assemblies of two 4S units and one turbine system. The capacity of the fuel cycle facility is 20ton/year and is able to reprocess the spent fuel from 4S for ten years.

The spent fuel discharged from the reactor after ten years of operation will be treated by IFR type reprocessing as advocated by ANL[6] so that it can be re-used as reactor fuel, while the long half-life wastes will be confined within the fuel cycle.

Thus, we could establish a self-supporting system in which plutonium is safely contained for a long time until more energy is needed, while covering the management cost with by revenues from power generation.

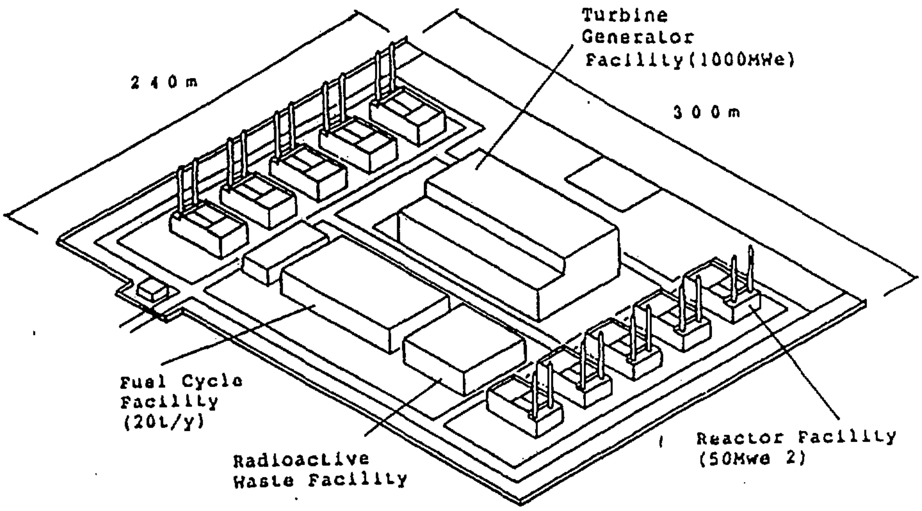


Fig. 19 Nuclear Power Plant Concept with 4S-Units

7. FURTHER R&D ITEMS

Although most of the technologies used in the 4S are already proven or under development, further R&D work is required for some key technologies. These are criticality experiments of the metallic core with reflector, higher reliability of the reflector driving mechanism and fuel performance of a long fuel slug.

A full scale critical experiment is important to evaluate the calculated results such as reactivity coefficients and critical conditions. As neutron leakage is enhanced in the 4S core, the conventional calculation method is not sufficient to accurately predict the core characteristics. A critical experiment is thus the most urgent R&D item.

All reactivity during plant operation is controlled just by moving the reflector without feedback control systems. Thus, a fine movements of the reflector are required. The technologies proposed for this purpose are all new to the nuclear industry, but are tried and tested in other fields. Reliability experiments are needed.

The feasibility of keeping a long metallic fuel slug in the core for ten years needs to be carefully examined. The preliminary assessment of creep deformation after ten years operation by the metallic fuel performance code shows about 5 % deformation and is within the allowable value. However, the performance of longer fuel slugs must be demonstrated.

8. CONCLUSIONS

Efforts to develop nuclear power reactors have been focused on the enhancement of safety and reliability by using active safety systems with redundancy and diversity while improving economy through scale factor. Although the goals have been fairly achieved, traditional power reactors can be used only under the prerequisites and are thus usable for a limited area of application.

Efforts have focused in recent years on developing small and medium sized power reactors with inherent, passive safety characteristics. Successful development of such reactors will lead to a wider range of application of nuclear power.

The 4S is a new concept of fast reactor designed to meet the goals of nuclear power and offers many attractive advantages. Commercial operation of the 4S is expected to solve a number of problems that humans will encounter in the 21st century.

Finally, the author would like to emphasize the need for worldwide cooperation on the development of the 4S reactor.

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USE OF NUCLEAR SPACE TECHNOLOGY OF DIRECT ENERGY CONVERSION FOR TERRESTRIAL APPLICATION

XA0056276

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Abstract

In due time the SSC RF-IPPE exercised the scientific supervision and directly participated in the development, fabrication, space flight test and maintenance of the direct energy conversion nuclear power plants (NPP) for space application under the „BUK“ and „TOPAZ“ programs. We have used the acquired experience and the high technologies developed for the „BUK“ NPP with a thermoelectric conversion of thermal (nuclear) energy into electrical one in the development under the order of RAO „GAZPROM“ of the natural gas fired self contained thermoelectric current sources (AIT-500) and heat and electricity sources (TEP-500). These are intended for electrochemical rust protection of gas pipelines and for the electricity and heat supply to the telemetric and microwave-link systems located along the gas pipelines.

Of special interest at the moment are the new developments of self contained current sources with the electrical output of ~500 Wel for new gas pipelines being constructed under the projects like the „Yamal – Europe“ project. The electrochemical rust protection of gas pipelines laying on unsettled and non-electrified territory of arctic regions of Russia is performed by means of the so-called Cathodic Protection Stations (CPS). Accounting for a complex of rather rigid requirements imposed by arctic operating conditions, the most attractive sources of the electricity supply to the CPS are the thermoelectric heat-into-electricity converters and the generators (TEG) on their basis. This paper deals with the essential results of the development, investigation and testing of unconventional TEGs using the low-temperature bismuth-tellurium thermoelectric batteries assembled together as tubular thermoelectric batteries with a radial ring geometry built into the gas-heated thermoelectric modules, which are collected to make up either the thermoelectric plants for heat and electricity supply or the self contained power sources. One of peculiarities of these plants is the combination of the CPS proper function with a number of additional service possibilities, among them the use of natural gas directly from the high pressure gas pipeline, the remote monitoring, the utilization of heat etc.

INTRODUCTION

To ensure effective cathodic protection (CP) of gas pipelines in the remote arctic regions of Russia shown in Fig.1, where it would be uneconomic to run power lines and where conventional techniques such as small gas or diesel generators, wind and solar power plants fail, it is possible to use thermoelectric converters of heat into electricity and based on them stand-alone thermoelectric generators (TEG).

Fig. 1. Map of power service in Russia.

Power service areas:
- centralized; - independent; - non - electrified.

Yet commercially available natural gas fuelled TEGs either possess small power (15-160 W el. - Russian market) or are less suited for arctic conditions (15-550 W el. - Canadian market) because:

- frequent starts cause a drop in electrical output power and reduce unit life;
- operation of non-Russian TEGs using the heat pipes' based heat removal system is only reliable where the power unit maintains its correct installation angle;
- control and electronic systems of the TEG matching to electrical load are inadequate resulting in seasonal over/undersupply of current to the cathodic protection station (CPS) because of seasonal variation in the load resistance (between the anode, the insulation on the gas pipeline and the gas pipeline proper);
- there is no remote monitoring function for arctic regions difficult of access;
- additional gas preparation plant is required to dry and preheat the pipeline natural gas;
- non-Russian plants are not adapted for Russian arctic conditions of operation on gas pipelines.

Apart from tackling these drawbacks, our development is aimed to bring down the high unit costs, improve efficiency and optimize the use of heat recovered for space heating on small technical premises, in residential buildings or camps in the field.

The new system jointly developed by SSC RF-IPPE (with its subsidiary "ECS-Russia), DAO "PROMGAZ" of RAO "GAZPROM" and "RUHRGAS AG" is based on a TEG design, which proved successful as a component of nuclear power plant and has to date the experience of stable operation for not less than 18 years with not less than 1300 thermal cycles (on/off). By now it is has been modernized by the development participants for use in natural gas heated systems and equipped with additional devices for the gas supply, the control over performance of power units and power/heat recovery system, the functioning remote control etc.

We believe that our CPS differs advantageously from the thermoelectric CPS available in Russian and world markets with the electrical output of 12-160 W (for example, of the GT-G-10-12 / 30-12 / 160-12 and other types fabricated in Russia by the company "Pravdinsky current sources' experiment plant" - POZIT) and of 15 to 550 W (for example, like 5015 / 5030 / 5060 / 5120 / 5220 / and 8550-24 models fabricated in Canada by the company "Global Thermoelectric").

This paper describes:

- the concept of new power system, which is geared to the climatic conditions of the arctic region of Russia;
- the optimization of the individual system components, such as the thermoelectric battery, the thermosiphon for heat recovery, the electronic system of matching to electrical load, the system of remote control over the power system functioning.

We developed two types of thermoelectric CPS: the TEP-500 type being the cogeneration thermoelectric system generating electrical and thermal energy for a Customer and the AIT-500 type being the stand-alone current source generating for a Customer electrical power only.

AIT-500 and TEC-500 performance data:

Fuel type	natural gas	
Fuel source	gas pipeline	
Inlet gas pressure, bar	up to	100
Fuel flow rate, nm^3/hour	no more than	3
Electrical power supplied to a load, W	not less than	500
Direct current voltage at output terminals, V		12...48
Heat power supplied by TEP-500 to a load, W		17000
Ambient temperature, K (°C)		223÷303(-50÷+30)
Allowable number of thermal cycles, pcs.	not less than	1000
Life time, years	not less than	10
Overall dimensions:		
AIT-500, m	0,9x0,9x3,5	
TEP-500, m	1,5x1,3x2,0	
Mass, kg	no more than	400

TEG-PLANT CONCEPT. SPECIAL CONFIGURATION FOR USE IN ARCTIC CONDITIONS

Developments have resulted in a novel TEG plant, which is suited for use in arctic regions and features various advantages over conventional TEG systems. The basic component of our TEG system is the thermoelectric module presented in Fig.2. It is unified for various types of TEG systems and differs for various plants in cooling system only. In cogeneration plants (of the TEP-500 type) the non-freezing heat transfer agent is pumped through a cooling system, whereas in stand-alone current sources (of the AIT-500 type) the cold junctions' cooling system is a part of convection system for dissipation of heat removed from the thermoelectric battery cold junctions' area. Fig.3 and 4 present functional diagrams of our TEP-500 and AIT-500 plants. Let us consider below the basic functional systems of these TEG plants.

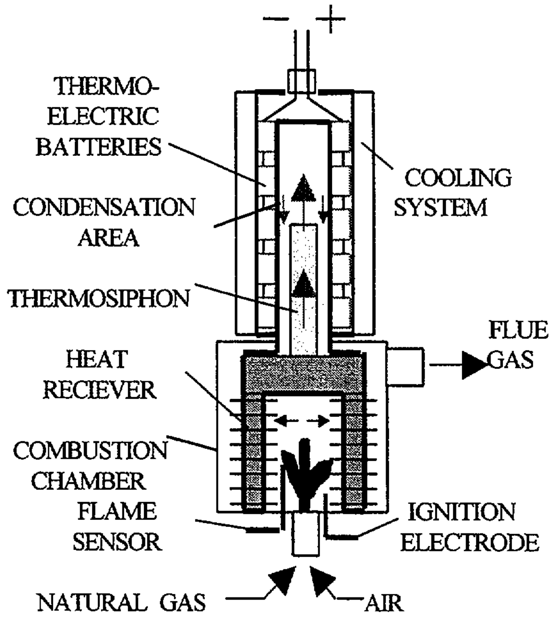

Fig. 2. Basic thermoelectric module (TM) schematic diagram.

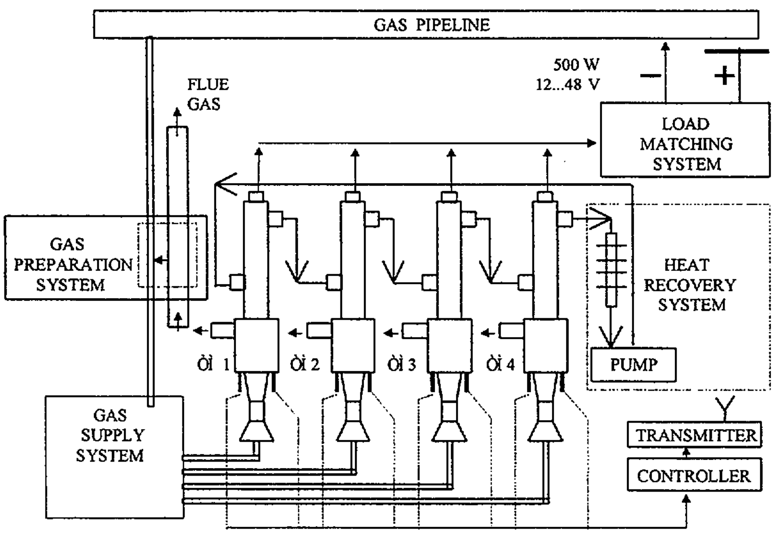

Fig. 3. TEP – 500 plant schematic diagram.

Thermoelectric batteries with long life time

The thermoelectric batteries consist of specially arranged pairs of semi-conductors made of bismuth-telluride alloys. While electrical efficiency of thermoelectric batteries is relatively low (~4%), the long thermocyclic life is ensured at low allowable hot junctions' temperatures of ~570 K owing to special geometry of thermoelectric batteries (the radial cylindrical one). Moreover, they can be produced in the CIS, which makes them a cost-effective component. Hence the tubular thermoelectric batteries are the basic component of a tubular thermoelectric module comprised of a needed number of tubular batteries connected in series.

Thermosiphons for optimum heat transfer and use

The specially adapted heat pipes (HP) / thermosiphons installed allow the good heat transfer characteristics typical of two-phase liquid-steam mixtures to be used for effective TEG

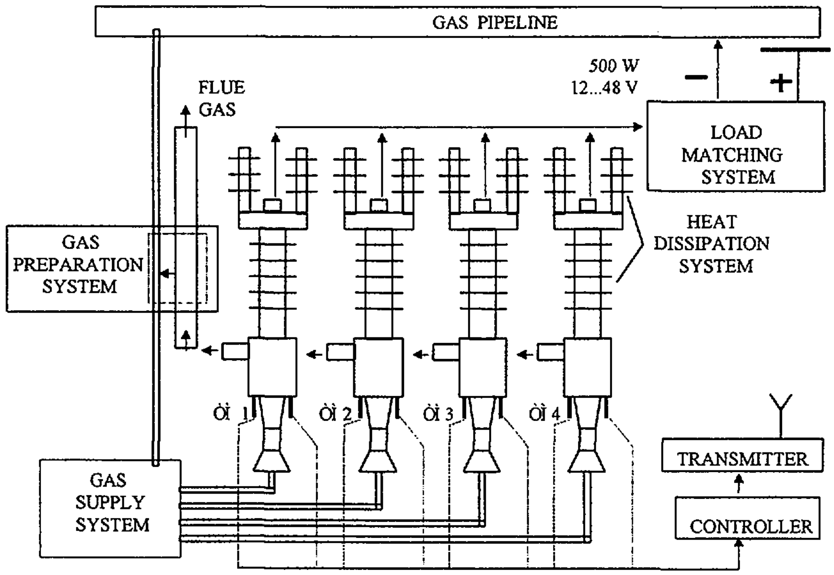

Fig. 4. AIT – 500 plant schematic diagram.

heating and cooling. The non-freezing HP working media assist to comply with specification on the plant efficiency at ambient temperatures of 223 K to 303 K (-50°C to +30°C).

Special features of fuel gas components for low ambient temperatures

In Europe, the use of gas control devices is limited to ambient temperatures down to 248 K (-25°C), in special cases down to 233 K (-40°C). For gas supply to our TEG plants, special sealing and membrane materials, steel bodies and noble contacts were purchased in the market and adapted for use in this harsh environment. Thanks to unconventional thermosiphon gas preheater built into the TEG plants and able to heat the gas supplied directly from the gas pipeline, either by flue gas heat or by means of small power electrical heater, which we have developed and fabricated, the TEG plant can be started and operated permanently at the natural gas inlet pressures between 4 and 100 bar without requiring neither secondary energy source nor additional gas preparation device.

Modular design for high operational safety and flexibility

The necessary electrical output to the load is provided by a total of four independent TEG modules. This warrants operational safety and gives considerable flexibility in adapting the output characteristics to the needs of CPS with varied electrical power demand by connecting the individual modules as required. Moreover, an open architecture of our TEG plant allows the flexible adjusting to various needs of a Customer and the connection of modules into parallel and series circuits contributing by this means to the reliability of a plant as a whole.

Heating circuit for heat recovery

The residual heat absorbed by the cooling system is removed by a fluid circulating in the closed heat exchanger system and dissipated via external radiators or heat exchangers. Circulation is by low-wear and low-loss electric pumps built into the cooling system. The electric pumps use a small part of the electric power generated by the plant. In case of stand-alone current sources, the heat dissipation system constitutes the tubular finned gas regulated heat exchanger (thermosiphon) automatically varying the area of convective heat exchange with ambient media under varying external climatic conditions.

Intelligent load matching system with remote monitoring

To avoid excessively high soil potentials and an undersupply of CP current, the TEG plant is equipped with a microprocessor-based load matching system, which automatically compensates seasonal variation in power demand by adapting the system's electrical output to match the load. The plant monitoring and control system is designed to give the main operating parameters as digital signals for the plant turning-on, the monitoring of its parameters and the shooting (and elimination) of possible troubles. The radio transmitter sends the plant status signal along with the TEG-plant identification signal to a central station intended to supervise various CPS operating along the gas pipeline (master station).

OPERATING EXPERIENCE

The correct operation of the above described functional units has been demonstrated in laboratory tests of individual TEG modules and TEG plant prototypes of the TEP-500 and AIT-500 types. The laboratory testing confirmed the validity of technical solutions selected in the process of the plants' development, as well as a good agreement with RAO "GAZPROM" requirements on operation in arctic conditions of Russia. Fig.5 and 6 present the TEP-500 and AIT-500 plants' general view photographs made in the course of laboratory testing.

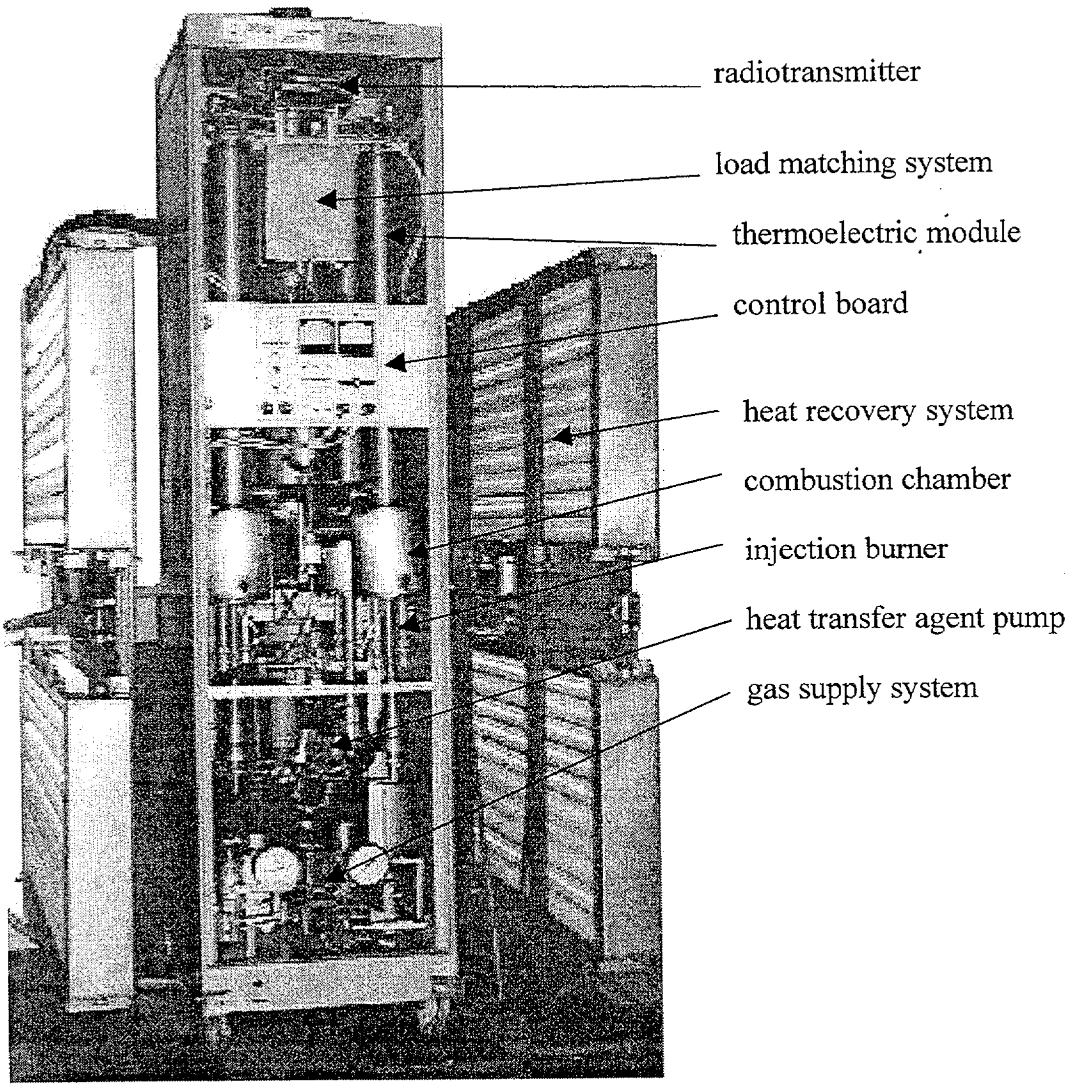

Fig. 5. TEP – 500 plant general view.

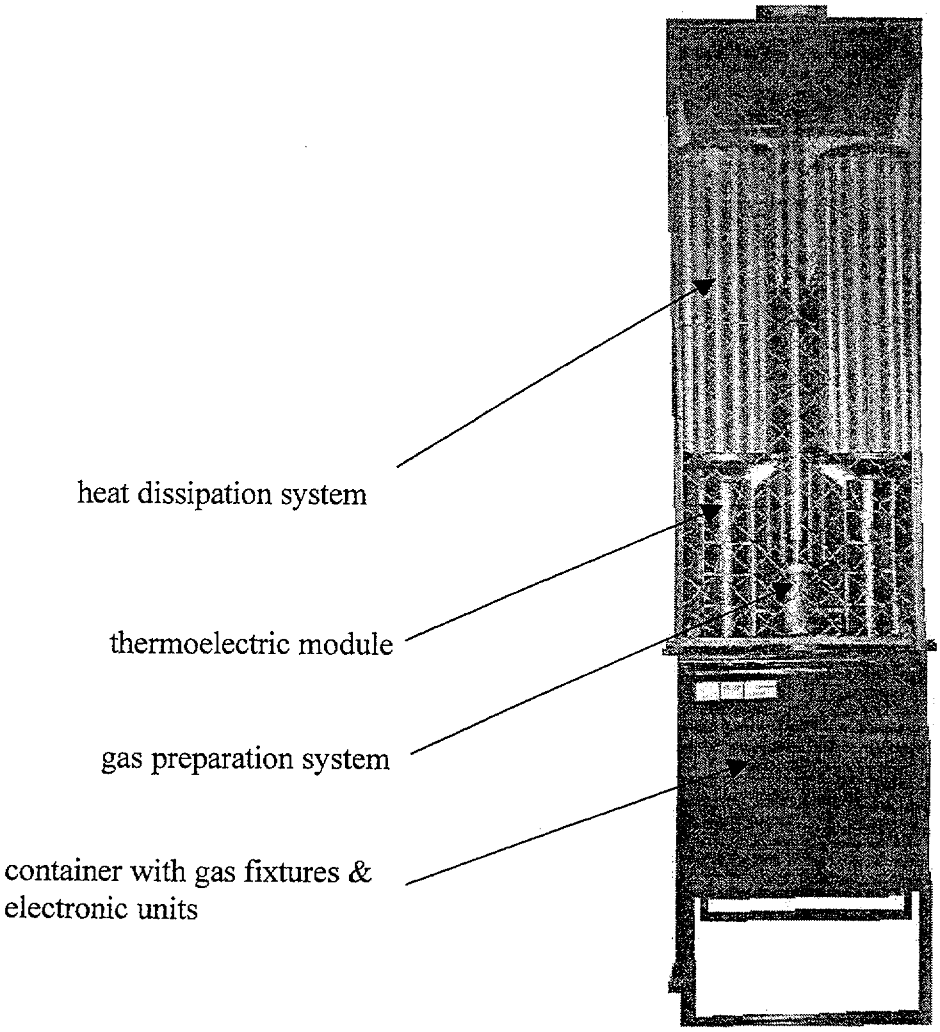

Fig. 6. AIT – 500 plant general view.

Starting from February 1998, the AIT-500 TEG plant has been put to field testing in one of subdivisions of "Lentransgaz" being the RAO "GAZPROM" subsidiary on the gas transportation and supply. The AIT-500 pilot operation to be carried in varying climatic and seasonal conditions, should demonstrate to the Customer that the plant meets the requirements of RAO "GAZPROM"in the conditions of real operation on the gas pipeline. For now, the results of field testing are positive. The TEP-500 plant is being prepared for field testing.

CONCLUSIONS

Thanks to special plant design and units used, our TEG plants offer the following technical advantages over commercial TEG systems available in the world market:

- great number of possible starts / long life time;
- suitability for arctic conditions;
- flexible variation of required electrical output; system adapted to seasonal changes in electricity demand;

- additional service functions for stand-alone CPS operated in regions difficult of access;
- optimum use of heat for indirect heating of natural gas from the gas pipeline and possibility of heat recovery;
- secure operation irrespective of place of installation and installation angle;
- series production of plants possible without outside manufacturer, apparent flexibility of plants in completing with standard components;
- plant costs are substantially lowered as compared to plants in the world market.

By this means, this design helps to overcome most of technical and economical obstacles facing TEGs where used in cathodic protection stations under arctic conditions. They are therefore an important contribution to effective cathodic protection of natural gas pipelines and hence to the security of gas supply for RAO "GAZPROM" and its customers.

DEVELOPMENT OF A NEUTRON THERAPY TREATMENT COMPLEX

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Abstract

Oncological diseases are still among the most common causes of untimely deaths. Neutron and neutron-capture therapy are rather effective methods at treatment of oncological diseases. Now these methods are practiced on expensive research reactors and accelerators , which were not projected for medical purposes. Therefore there is the problem of design of special reactor installation, intended for radiation therapy methods.

Such installation should provide necessary density of neutron beams, it should be safe and inexpensive.

In this paper two samples of irradiation installation for neutron and neutron-capture therapy are submitted. The first irradiation installation is based on application of the water-water "pool" type reactor by power 110kW , the second - on application of liquid-metal Na-K coolant reactor by power 40kW with zirconium hydride moderator.

The irradiation unit for both reactors has three channels: for fast neutron therapy beams, for neutron-capture therapy using thermal neutron beams, for therapy using mixed gamma-neutron beams. The cost of both installations low - about one million dollars.

1. INTRODUCTION

Oncological diseases are still among the most common causes of untimely deaths in industrially advanced countries. A large percent of oncological patients need radiotherapy, the progress in which essentially depends on achievements in nuclear industry.

One of the most promising radiation therapy directions is connected with densely ionizing radiation. Recently much convincing evidence of successful neutron application in malignant tumour radiation therapy have been obtained in USA, Russia, England, Germany, France, Japan and other countries. It should be noted that the beginning of fission-neutron treatment began in Obninsk at the end of 1960's with experimental and clinical research carried out by both the Medical Radiology Science Centre (MRSC) of the Russian Academy of Medical Science and the Institute of Physics and Power Engineering (IPPE) using the BR-10 reactor. During these studies, about 230 patients with widespread malignant tumours were cured, among them 139 neuro carvical and facial carcinoma, 59 thoracic carcinoma and 20 sarcoma. The efficiency of neutron therapy methods is high. Survival rates 3+ years for cancer of the head and neck was increased from 68% (for gamma therapy) to 94% (for neutron therapy). For other cancers, positive results were also achieved and they are now in a stage of statistical processing.

To date, the following methods have been recognized to be promising:

a) boron neutron therapy (BNCT) (USA, Japan, Great Britain, Germany), connected with the appearance of a new generation of tumorotropic boron containing compounds;
b) beam therapy in mixed gamma-neutron fluxes (Russia);
c) radiotherapy with fast neutron beams.

As a rule, neutron therapy uses neutrons generated by very expensive accelerators and cyclotrons. To obtain neutron fluxes of greater densities, i.e. about 10^{10} n/cm^2×s, research reactors are used. However, the use such reactors for such medical applications entails serious difficulties, including:

- these reactors were constructed without provision for medical requirements;
- they are complex and very expensive;
- as a rule, other ongoing technological works are carried out on them, so treatment plans have to be adapted to operation modes.

Thus, it may be concluded that it is necessary to create a special basic medical reactor (SBMR) which is simple, reliable and inexpensive for medical application only. This would be provide a wide use of neutron therapy methods.

IPPE has wide experience and technologies of creation of various types reactors, since reactor by World's First NPP with the water coolant and finishing space reactor of a type "TOPAZ" with (Na-K) -coolant. Therefore was decided to consider a possibility of creation SBMR on basis of being available technologies and experience of operation of acting reactors.

2. REQUIREMENTS AND SELECTION CRITERIA FOR AN SBMR

The authors were guided by the following basic requirements when selecting features of the SBMR conceptual design:

- to meet physician requirements for neutron beam densities – of 10^8 to 10^9 n/cm^2 ×s for fast neutrons and about 10^{10} n/cm^2 ×s for thermal neutrons;
- to provide necessary values of neutron beam densities with the least divergence at the lowest reactor thermal power:
- to provide a reactor resource within 10÷15 years;
- to use available technologies developed by IPPE for construction materials and coolants;
- to use convection heat removal;
- to obtain a small cost per irradiation unit with the SBMR

The problem of creating such a reactor is actually two fold:

a) selection of the reactor type, and
b) design and development of efficient collimator systems placed into a biological shield.

When selecting the reactor type for radiation therapy one must start from a principle that is exactly opposite to general practice. Ordinarily reactor designers try to return all the neutrons into the core, thus reducing the reactor dimension and the loading of fissile materials in it. However, for a medical reactor, the problem is to provide maximum emission of neutrons from the reflector surface. So, the part of outcoming neutrons per one fission is an important parameter for reactor type selection.

Another selection parameter is the core size, since a smaller core provides a larger leakage of neutrons and, in addition, allows the formation of neutron beams with smaller divergence.

Collimator selection is the key issue for radiation therapy. We selected a conical configuration, which makes a possible to reduce reactor power by orders of magnitude while still providing: a) the same densities of neutron fluxes as those in research reactors, and b) normal dosimetrical conditions around the reactor.

Materials selection for the collimator system is an important problem, as it must provide an efficient means to return neutrons to the beam, slow down neutrons in a channel for neutron-capture therapy, or multiply them by means of converter for fast neutron therapy. It is worth nothing that the collimator problem is the same for any reactor type.

Taking into account all the requirements and criteria for reactor type and collimator selection, we arrived at two potential: a) a water-water "pool"-type reactor, and b) a reactor with liquid-metal Na-K coolant. Both variants are simple to control, have increased reliability and safety, the heat is removed by natural convection without the use of pumps. Manufacturing costs are very low.

3. REACTOR INSTALLATION WITH LIQUID-METAL COOLANT

The set-up and a table of main parameters of this version are presented in Fig. 1 and Table I respectively.

Table I. Base parameters of SBMR.

Parameter	Dimension	Value Water-water "pool" type reactor	Reactor with Na-K coolant
N_0	kW	110	40
G^5	kg	3.6	5
T	year	15÷20	15÷20
$D_{core} \times H_{core}$	cm	25×24	25×25
$S_{reflector}$	cm	12	12
$d_{fuel\ rod}$	cm	0.91	0.91
$N_{fuel\ rod}$	unit	123	100
$N_{cylinder}$	unit	8	8
$\Delta K_{control\ rod}$	$\%k_{eff}$	~1.1	~1
$\Delta K_{temperature}$	$\%k_{eff}$	-6	-3
ΔK_{H2O}	$\%k_{eff}$	—	-0.1
ΔK_{void}	$\%k_{eff}$	-15	-0.5
T_{input}^{Na-K}	^0C	—	360
T_{max}^{Na-K}	^0C	—	400
T_{input}^{water}	^0C	80	—
T_{max}^{water}	^0C	90	—
P_{core}	at	~1	~1
T_{max}^{air}	^0C	—	190
neutron leakage	n/fission	~0.8	~1
n_{beams}	unit	3	3
$\varphi_{fast\ neutron}$	$1/cm^2 \times s$	3×10^9	3×10^9
$\varphi_{epithermal\ neutron}$	$1/cm^2 \times s$	4×10^9	4×10^9
D×H	$1/cm^2 \times s$	4×6	4.7×6
H_{stack}	m	—	10÷15
cost	million USD	~1	~1

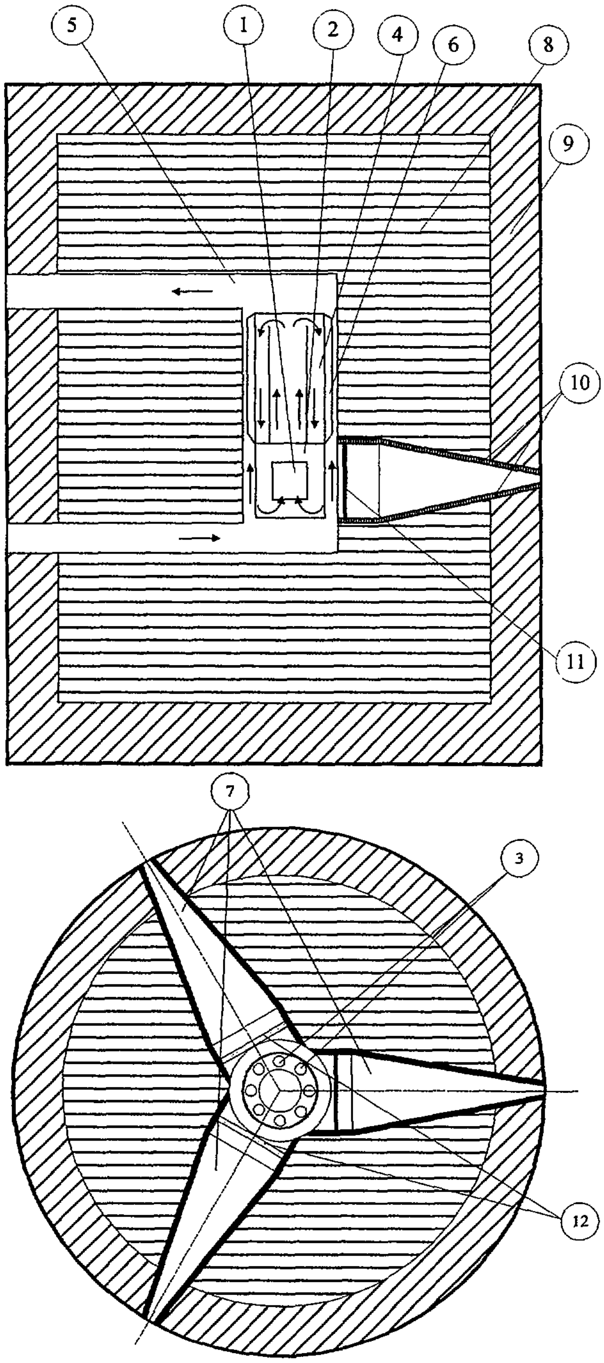

Figure 1. Scheme of "liquid metal" SBMR.
1. Core
2. Reflector
3. Control rods
4. Na-K circuit
5. Air circuit
6. Heat removal ribs
7. Channels of beams
8. Light boron concrete
9. Heavy concrete
10. Reflector coating
11. Converter
12. Filters

Calculations show that using a uranium dioxide fuel with 60% enrichment on U^{235} with a zirconium hydride moderator makes it possible to obtain a core size of no more than 25 cm. The reactor is controlled by rotary steel drums with insertion from boron carbide, all drums are located in the reflector. The reactor shielding has a total thickness of 2 m, consisting of two layers: light (1.6m) and heavy (0.4m) concrete.

Collimators are placed as it shown in Fig. 1 They have a conical configuration with base sizes that equal the reactor ones. Collimator walls are lead with a layer thickness 0.1m. Different kinds of moderators, filters and converters can be used in the collimators.

Convection heat removal using two-planimetric circuit is proposed for this version. The first contour – liquid-metal – is located completely in the reactor vessel and works under Fild's scheme. The coolant, heated in the core up to 400^0C, rises upwards alone the central part of the reactor, turns and descends at a wall the along periphery, having been cooled to 360^0C. The second contour is the air. Heat is removed from the ribbed surface of the reactor vessel by means of the air, which is heated up to 180^0C and then exhausted through the stack.

The use of the (Na-K) coolant in the first contour, with maximum temperature $<400^0$C makes it possible to have safe normal pressure inside the reactor vessel. All structures and the configuration of the core were chosen to provide maximum nuclear and radiation safety.

4. INSTALLATION WITH WATER-WATER REACTOR

A diagram and a table of the main parameters of the water-water version are presented in Fig. 2 and Table I respectively. The same fuel rods are used as in the first version. In this case, water plays a multifunctional role as a moderator of neutrons, as a coolant, as an accumulator of allocated heat and a biological shield. The two-layers Be-steel screen serves as a reflector.

The reactor is controlled by the same rotary drums as in the first version. The total thickness of the biological shield is less and equals 1.75m. It consist of water layer in the pool (1.25m), a steel layer (0.4m) and an external layer of boron polyethylene (0.1m). Collimators are similar those of the first version. The heat, generated by the reactor is absorbed by the whole weight of the pool water, thanks to its high thermal capacity. During continuous 8-hour reactor operation the temperature effects no more then 8^0C. All reactor effects, connected with change in its condition (temperature, void) have negative values.

Comparing the date in Table I it is clear that the power of a pool reactor must be increased up to 110kW to provide the same density of neutron beams. This is due to decreased neutron emission in the water-water reactor, a smaller effective scattering surface near the reactor due to the absence of a cavity between the reactor and shield, and finally, due to worse albedo properties of water compared to those of concrete.

5. THE OPERATIONAL AND CONSUME CHARACTERISTICS OF THE IRRADIATION UNIT

The irradiation unit for both reactors has three channels: 1) for fast neutron therapy beams (3×10^9 n/cm$^2\times$s), 2) for neutron-capture therapy using thermal neutron beams (4×10^9 n/cm$^2\times$s) and 3) for therapy using mixed gamma-neutron beams.

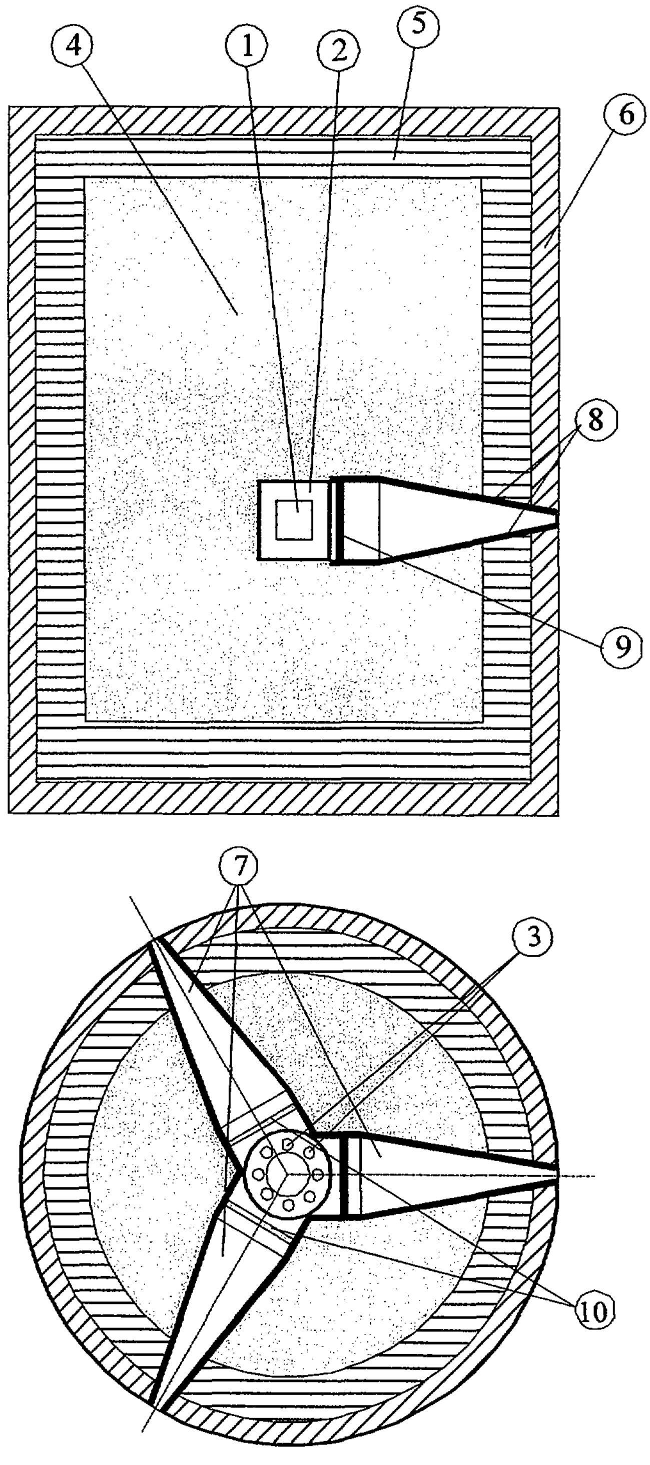

Figure 2. Scheme of "pool"- type SBMR.

1. Core
2. Reflector
3. Control rods
4. Water
5. Steel
6. Boron polyethylene
7. Channels of beams
8. Reflector coating
9. Converter 10. Filter

The irradiation unit is designed for 15÷20 year lifetime without intermediate refueling. Due to the convection method of heat removal and its rather small dimensions, the unit is simple to control and does not require large expenses for its construction. The staff is small – approximately 5 people without medical staff. An eight-hour work day is planned. The duration of one session, with all beams functioning, equals 20 minutes, with a pure irradiation time of no more than 5 minutes. The reactor can be switched on and off several times during the work day. The dose in adjacent rooms does not exceed extremely acceptable value of 20mZv/year for professional staff. Selection of reactor parameters and constructional materials will prevent the possibility that fuel rods could melt in hypothetical accidents. Cost estimates for the irradiation unit are < million US dollars for the reactor types considered.

On the basis of the irradiation unit described above, it is possible to create a treatment complex with a clinic for 130 patients. As the treatment course consists of 10 session for month, an annual possible number of patients to treat is 1600. It is known that for 1 million people there are 3 thousand oncological cases, and 10% of these cases are effectively treated by fast neutron therapy. So, in order to satisfy the needs of medicine in this direction, for example, for Moscow two treatment complexes would be needed, and for Russia in the whole about thirty.

Implementation of this project for wide use of neutron therapy would save tens of thousands of lives annually.

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